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GENERAL

9575 CF-56-7-157
Oak Ridge National Lab., Tenn.
REPORT OF EXPLOSION AND FIRE IN HYDROGEN LIQUEFIER APPARATUS. T. W. Hungerford, J. E. Lain, and B. M. Beeler. July 31, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

BIOLOGY AND MEDICINE

9576 UCLA-362 (Del.)
California. Univ., Los Angeles. Atomic Energy Project.
QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1956. Apr. 1, 1956. Decl. with deletions Mar. 4, 1957. 104p. Contract AT-04-1-GEN-12. $16.80 (ph OTS); $5.70(mf OTS).

Progress is reported in the following studies: the synthesis of labeled fatty acids; the pharmacological properties of a series of oxazolium compounds in mice; the effects of atropine on the response of rat uterus to 5-hydroxytryptamine and acetylcholine; the effect of quinoxaline 1:4 di-n-oxide on survival time of irradiated mice; the effect of acute and chronic intravenous infusions of clinical dextran in irradiated rabbits; the effect of particle size on pulmonary absorption and distribution; the development of kidney function tests employing radioactive Diodrast and external gamma ray scintillation counting techniques; blood radioactivity measurements; the mechanism of the skeletal deposition of Ca^{45} and Sr^{90}; the production, extraction, and purification of an antibiotic substance produced by an intestinal lactobacillus; the late effects of total-body x irradiation on rats exposed under 5% hypoxia; radioinduced lesions in the eye of the newly metamorphosed toad; routine monitoring and health physics procedures; the development of procedures for the spectrographic analysis of bone ash; design of a freeze-dryer for the preparation of tissue specimens for electron microscopy; the production and calibration of chemical systems of radiation measurement having medical applications; the physiological effects of air blast in mice; tracer studies on the plant uptake of Sr and Cs from various soil types; tracer studies on the effect of Ca on the uptake of Sr by plants and the uptake of Sr from plant material ingested by animals. (C.H.)

RADIATION HAZARDS AND PROTECTION

9577 KAPL-814
Knolls Atomic Power Lab., Schenectady, N. Y.
RADIOLOGICAL DEVELOPMENT ACTIVITIES. Health and Safety Unit Semiannual Progress Report for January–June 1952. July 1952. Decl. Mar. 29, 1957. 42p. Contract W-31-109-Eng-52. $7.80(ph OTS); $3.30(mf OTS).

Data are reported from a study of the total activity, the size distribution of the particulate material, and isotopic composition of the radioactive components discharged from the stack during the dissolving, head-end, and extraction operations of the separations process. The atmospheric dilution of the stack effluent was also estimated. Information obtained on the radioisotopic composition of evaporator condensate provided the basis for revision of the maximum permissible limits for discharge to the Mohawk River. A series of tests established a maximum detection efficiency of 19% for a scintillation probe in detecting γ radiation in a liquid medium. A smoke generator suitable for the evaluation of the efficiency of laboratory fume hoods is described and a schematic sketch included. (C.H.)

CHEMISTRY

9578 ANL-4593
Argonne National Lab., Lemont, Ill.
SUMMARY REPORT FOR OCTOBER, NOVEMBER, AND DECEMBER, 1950; CHEMISTRY DIVISION, SECTION C-1. D. W. Osborne, ed. Feb. 1, 1951. Decl. Feb. 14, 1957. 48p. Contract W-31-109-eng-38. $0.45(OTS).

The threshold for the photoneutron reaction $U^{238}(\gamma,n)U^{237}$ and the U^{238} fast-neutron fission and capture cross sections are reported. The yield of 6.6-day U^{237} vs. x-ray energy is plotted. The measurement of the flux above the U^{238} threshold data n the variation of U^{238} fission cross section with neutron energy, and data on the distribution of neutrons with respect to energy are combined to calculate the total fission neutron flux. In basic chemistry the following studies are reported: preparation of PaF_4 from PaO_2, preparation of Pa from PaF_4, properties of Pa, potentiometric measurements for thermodynamic properties of the reaction $Np^{5+} + H^+ \rightarrow Np^{6+} + \frac{1}{2}H_2$, thermal decomposition products of $UO_2C_2O_4$, reactions of U, Pu, and $Pu(IO_3)_4$ with BrF_3 plus Br, HF, KF, or SbF_5, distillation equipment for BrF_3 and solubility of metal fluorides in BrF_3. Other investigations reported are concerned with dissociation pressures of UH_3, UH_3-UT_3, and UT_3, the preparation of T-labeled $LiBH_4$, behavior of rare earths in TBP solvent extraction processes, chemistry of Re, heats of solution of $Co(NO_3)_2$ hydrates in various organic solvents, adsorption of $CoCl_2$ from organic solutions by anion exchange resins, electrochromatographic separation, and solidification of He^3. (T.R.H.)

9579 ANL-5000(Del.)
Argonne National Lab., Lemont, Ill.
CHEMISTRY DIVISION, SECTION C-II SUMMARY REPORT FOR JULY, AUGUST, AND SEPTEMBER 1952. J. R. Gilbreath and O. C. Simpson, comps. Jan. 21, 1953. Decl. with deletions Feb. 12, 1957. 75p. Contract W-31-109-eng-38. $12.30(ph OTS); $4.50(mf OTS).

The progress is reported of the work on the physical properties of graphite; effects of reactor radiation on the properties of graphite; effects of irradiation on ceramic materials; x-ray-induced luminescence of ice; investigation of color centers and other optical properties of single crystals; radiation chemistry of liquids; application of mass spectrometry to studies of isotopic exchanges of O; vapor pressure and heat of vaporization of U; nuclear properties of Zr^{93} and Nb^{93m} from fission; mass distribution in the spontaneous fission of Cm^{242}; upper limit to lifetimes of first excited states of Th^{230}, U^{234} and Pu^{238}; spectrographic

and chemical analysis; and design and performance of the 60-in. cyclotron. (J.E.D.)

9580 ANL-5494(Del.)
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] JULY, AUGUST, AND SEPTEMBER 1955. Nov. 2, 1955. Decl. with deletions Feb. 5, 1957. 117p. Contract W-31-109-eng-38. $18.30(ph OTS); $6.00(mf OTS).

Progress is reported on the following investigations: entrainment of dissolved radioactivity from a boiling water surface; dispersion and coalescence in immiscible liquid systems; fused salt-fluoride volatility process for U–Zr fuel materials including dissolution of U–Zr and Sn–Zr in fused $NaF-ZrF_4$; design of shielded, vacuum-casting U furnace; dissolution of fissium as stand-in for EBR-II fuel elements; liquid Mg extraction of Pu; corrosive effects of Mg on Fe alloys; spontaneous ignition of U metal; Mo determination by solvent extraction of the xanthate into $CaCl_2$. (T.R.H.)

9581 ANL-5633
Argonne National Lab., Lemont, Ill.
CHEMICAL ENGINEERING DIVISION SUMMARY REPORT [FOR] JULY, AUGUST, AND SEPTEMBER 1956. Dec. 1956. Decl. Mar. 14, 1957. 147p. Contract W-31-109-eng-38. $0.80(OTS).

Solvent Extraction: Additional runs have been made in the six-inch, continuous-flow mixing chamber to study the rate of mass transfer between isobutanol and H_2O. Fluoride Volatilization Separations Process: Work has continued on the processing of Zr–U fuel alloys in fused fluoride systems, corrosion tests on graphite, behavior of Pu during U volatilization step of the fused salt process, crude UF_4 produced from ore concentrates by hydrogenation and hydrofluorination in fluidized bed reactors, and separation of MoF_6 from UF_6 by distillation. Kinetic data were obtained on the gas-phase reaction between F_2 and BF_3 to produce BF_5, and the separation of Mo and V from UF_6. Fluidization: Corrosion tests were made during the reduction and hydrofluorination of several ore concentrates. Installation of the green salt pilot plant was nearly completed. Fabrication of the fluidized-bed calciner for reactor fuel processing wastes was completed. Reactor Chemistry. The absolute fission yield of Cs^{137} has been determined for fast neutron fission of U^{235} and Pu^{239}. Samples of U^{235}, Pu^{239}, and U^{238} have been exposed in various parts of EBR-1 in order to determine the variation of the ratio of capture-to-fission cross sections, with reactor position. The study of oxidation characteristics of U, Th, Zr, and Pu has been continued. Chemical-Metallurgical Separation Processes: The corrosion of alumina, thoria, and beryllia is discussed. The extraction of Pu from hydride-produced 1% Pu–U powder by molten Mg is presented. Data have been obtained for the partition of Ru between liquid Mg and liquid U–Fe alloy. Testing of pilot plant equipment components was continued. Analytical Research and Services: An x-ray spectrometric method is being developed for rapid determination of Pu in concentration range 3 to 9×10^{-3} M. (For preceding period see ANL-5602.) (W.L.H.)

9582 CF-56-10-8
Oak Ridge National Lab., Tenn.
PREPARATION OF STANDARD OXIDE. R. B. Gallaher and A. S. Kitzes. Oct. 3, 1956. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

In order to facilitate correlation of data obtained by various groups studying homogeneous reactor slurries, the method of slurry preparation was standardized. Oxide preparation and slurry mixing and aging are outlined. (D.E.B.)

9583 MCW-196
Mallinckrodt Chemical Works, St. Louis.
SOLUBILITY OF PHOSPHATE IN RAFFINATE LIQUORS OF VARIOUS COMPOSITIONS. C. W. Kuhlman and C. F. Ritchie. July 7, 1949. Decl. Feb. 27, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

The quantity of phosphate which could be tolerated in typical plant liquors without incurring a precipitation of a uranium–phosphorus compound in any part of the plant was determined. The solubility of phosphate was determined in: (a) plant 6 raffinate; (b) raffinate from low grade CX feed material, (c) raffinate from high grade CX feed material. In (a) the concentration was varied in the range 0 to 260 g/l and the HNO_3 concentration was varied between 0 to 150 g/l. In (b) and (c), HNO_3 alone was varied. A solid phase containing Cu, presumably as $Cu(UO_2)_2(PO_4)_2 \cdot XH_2O$ was found to be in equilibrium with all the raffinates containing Cu when $UO_2HPO_4 \cdot 4H_2O$ was added as the solid phase. The maximum quantity of phosphate that could be handled by the existing process without loss of U by precipitation is 1.53% phosphate on a U basis. If the amount of Fe were increased in the ore to 16% of the U content, this maximum phosphate limit becomes 3.3% with respect to U. Appended is an account of a synthetic preparation of copper uranyl phosphate, $Cu(UO_2)_2(PO_4)_2 \cdot 8H_2O$. (A.M.)

9584 MonC-390
Clinton Labs., Oak Ridge, Tenn.
FIXATION OF NITROGEN BY PILE RADIATION AND ACID FORMATION IN THE ARGONNE PILE. A. O. Allen, T. W. Davis, and C. J. Hochanadel. Oct. 24, 1947. Decl. Feb. 28, 1957. 14p. Contract W35-058-Eng-71. $0.25 (OTS).

Nitrogen and O_2 combine in the gas phase under pile radiation to a product which will dissolve in water with formation of HNO_3. Little or no acid formation occurs directly in the water by action of radiation on dissolved N_2. Ammonium ion dissolved in water is slowly oxidized to HNO_3 by pile radiation. Acid which forms in the Argonne Pile moderator is neutralized with ammonia; the oxidation of the ammonia is responsible for the continuing formation of acid in the pile. The rate of acid formation in the pile is at present somewhat greater than expected from radiation effects, and it may be that ammonium ion is being oxidized in the pile also by some other process. (auth)

9585 NYO-1235
Hooker Electrochemical Co., Niagara Falls, N. Y.
THE VAPOR PHASE CONVERSION OF BORON TRIFLUORIDE TO BORON TRICHLORIDE. Anthony Loverde. May 24, 1954. Decl. Mar. 12, 1957. 15p. Contract AT(30-1)-1524. $3.30(ph OTS); $2.40(mf OTS).

Because the vapor phase conversion of BF_3 to BCl_3 offered many advantages, its possibilities were investigated. It was found that the reaction took place readily at temperatures over 225°C. Twenty seven runs were made in five different reactors. When an excess of $AlCl_3$ is used, yields averaging 97.5% of theory were obtained of product of 100% purity. (auth)

9586 NYO-1255
Hooker Electrochemical Co., Niagara Falls, N. Y.
THE RECOVERY OF BORON TRICHLORIDE FROM HYDROGEN, HYDROGEN CHLORIDE AND BORON TRICHLORIDE MIXTURES. E. A. Belmore and J. A. Spina. Aug. 15, 1955. Decl. Mar. 1, 1957. 50p. Contract AT(30-1)-1524). $7.80(ph OTS); $3.30(mf OTS).

In the process being operated for the reduction of BCl_3 to the metal, the effluent gases from the alloy wire reducer contained about 88% H, 9% HCl and 3% BCl_3. To recover the excess BCl_3 it, together with some of the HCl, was frozen out with liquid N. This work was carried out to develope a method of recovery, less expensive than the

liquid N one. "Molecular Sieves" and supported anhydrous $CuSO_4$ were screened as direct adsorbents for the HCl and activated C for the BCl_3. Since the C seemed the most promising the work was concentrated on this. Certain types of carbons such as the "Columbia" CXA 4 × 6 pelletts were found to adsorb as much as 40 to 45% of their own weight of BCl_3 with practically 100% efficiency up until the point that saturation is almost reached. Also it may be used for a large number of cycles if anhydrous conditions are maintained. The principal draw back for the use of the carbon for recovering this material in its permanent hold-up of about 10 to 15%. It was found that this could be effectively reduced by various treatments such as prechlorinating. A substantial saving over the liquid N method was indicated. (auth)

9587 ORNL-1674

Oak Ridge National Lab., Tenn.

CHEMISTRY DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING DECEMBER 20, 1953. Apr. 9, 1954. Decl. Mar. 5, 1957. 118p. Contract W-7405-eng-26. $0.70(OTS).

Progress is reported on the following investigations: anion exchange studies of the separation of weak acids from strong acids, adsorption of Os^{4+} from chloride solutions, adsorption of U^{6+}, Pb^{2+}, and Bi^{3+} from nitrate solutions, and separation of alkaline earths and alkali metals; hydrolytic polymerization of Zr^{4+}, Hf^{4+}, and Bi^{3+}; interaction of thiocyanate cations with Ag^{+} and Hg^{2+}; effects of low concentrations of foreign ions on the electrode potential of stainless steel; effectiveness of pertechnetate, perrhenate, and W ions as corrosive inhibitors; decomposition of Li amalgam; electrical conductivity and density of fused salts; miscibility of liquid metals and salts; half-life measurements of Ce^{144}, Co^{60}, and Kr^{85}; emission spectra and half life of Pm^{145}; radiochemical decontamination of porcelain enamel; mechanism of the Wagner-Meerwein rearrangement in the 1,2,2-triphenylethyl system; organic chemistry of solvents; crystal structure of oxalic acid dihydrate and iodic acid; Zeeman studies of the quadrupole spectra of Cl^{35} in $NaClO_3$ and $KClO_3$; thermodynamic properties of CdI_2; self-diffusion of cations in polystyrenesulfonate solutions; fundamental studies on ion-exchange selectivity; α-induced combination of H_2 and O_2 in the presence of CO_2; effect of temperature on decomposition of aqueous solutions by γ radiation; effect of fast electrons on corrosion of 347 stainless steel by UO_2SO_4 solutions; oxidation of U^{4+} by γ rays; design of pulse analyzers for various uses; decoding matrix for a fast binary scalar; cathodic protection of 309 SNb stainless steel exposed to a $Th(NO_3)_4-Al(NO_3)_3-HNO_3-HF$ mixture; phase studies of the $UO_2-SO_3-H_2O$ system, the $UO_2SO_4-NiSO_4-H_2O$ system, and $Th(NO_3)_4-HNO_3-H_2O$ system; estimation of activity coefficients of UO_2SO_4 solutions from vapor pressure measurements; corrosion of stainless steel and titanium by reactor solutions; and mechanism of the H_2-O_2 recombination reaction in solutions. (For preceding period see ORNL-1587.) (J.S.R.)

9588 ORNL-1931

Oak Ridge National Lab., Tenn.

STATISTICAL EVALUATION OF METHODS FOR THE ANALYSIS OF DIBASIC ALUMINUM NITRATE (DIBAN). J. G. Surak and P. F. Thomason. H. P. Raaen, ed. Sept. 16, 1955. Decl. Mar. 2, 1957. 41p. Contract W-7405-eng-26. $0.30(OTS).

The indicated methods for determining the following constituents of Diban, which is an aqueous solution of dibasic aluminum nitrate, $Al(OH)_2NO_3$, were evaluated statistically: aluminum by gravimetric, volumetric, and spectrophotometric procedures; basicity (hydroxyl value) by formation of an aluminum complex and titration of the free acid with standard alkali solution; total nitrogen by the Kjeldahl method; ammonia by the Kjeldahl method; and nitrates by means of a cation-exchange resin and titration of the liberated acid with standard alkali solution. Recommendations are made regarding the preferred methods of determining the constituents in $Al(OH)_2NO_3$ and regarding means of minimizing errors in these analyses. (auth)

9589 TID-10155

New Brunswick Lab., AEC, N. J.

QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING MARCH 31, 1952. C. J. Rodden. Sept. 12, 1952. Decl. Mar. 6, 1957. 121p. $0.70(OTS).

The tributyl phosphate extraction of thorium nitrate using a glass mixer settler is described. The results obtained on the extraction of a neutral aqueous feed with an acidified solvent are summarized. Distribution data from the mixer settler runs are also plotted. The extraction of solid thorium nitrate using an acidified organic feed is also described, and results are tabulated. A mixer-settler apparatus, used for the concentration of impurities after extraction, is shown, and the concentration processes are described. In order to determine a procedure for the solubilizing of a mixture of thorium and rare earth oxalate for solvent extraction, a study was made of the thermal decomposition of thorium and lanthanum oxalate. The rate of pyrohydrolysis of ThF_4 at 500°C was studied, and its application to ThF_4 sludges is discussed. The distributions of yttrium and ytterbium between TBP and thorium nitrate–nitric acid solutions are given, as is the distribution of nitric acid in systems containing 30% TBP in kerosene and thorium nitrate. Preliminary data on the thermal decomposition of uranyl nitrate hexahydrate are given. Results on the separation of Th from the rare earths by peroxide precipitation are also given. Salt bath contamination of slugs was studied. Methods for the precise determination of radium are discussed. A radioactivation analysis for indium is also described. The application of an induction heater to the determination of C in U metal is discussed. Complete designs are given for a cathode ray tube microdensitometer. (B.J.H.)

9590 TID-10158

New Brunswick Lab., AEC, N. J.

QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING JUNE 30, 1953. C. J. Rodden. Decl. Mar. 7, 1957. 66p. $0.50(OTS).

Potassium fluoborate can be prepared from the dimethyl ether-boron trifluoride complex by the reaction of the complex with an alcohol-aqueous solution of KF. The salt can be rapidly filtered and dried. Products of high purity, with yields up to 99.8% were obtained. H_3BO_4 can be prepared from dimethyl ether-boron trifluoride by the reaction of an aqueous solution of the complex with $CaCO_3$. Filtration of the CaF_2 is rapid when a filter aid is employed. H_3BO_4 of high purity can be obtained when crystallized from the filtrate at 0°C and washed with an acetone-ether mixture to remove organic matter. Yields up to 98.0% as H_3BO_4 and 94.6% as boric oxide were obtained. High enrichments of U^{235} were determined by the measurement of fission product gamma rays caused by neutron bombardment of a sample. Concentrations of U^{235} up to 93.3% were determined by irradiating the sample in a Ra–Be neutron source for ten minutes and counting the fission product gamma rays. Counting rates higher than 100,000 counts per minute were obtained with a NaI(Tl) crystal scintillation counter and fifteen minutes were required for a determination. The corrected gamma ray counting rate was a linear function of the weight of U^{235} irradiated. Using the method described, an accuracy of 0.5% was obtained. A spectrophotometric method for the estimation of microgram quantities of Th in U metal is presented. The procedure involves the solution of the metal in HNO_3 and the

separation of the Th from the U by coprecipitation with Hg as the iodate. The iodates were dissolved in HCl and reduced with $NaNO_2$ to the chlorides and free iodine. The I was removed from the chloride solution by extraction with methyl ethyl ketone. The determination was concluded by a spectrophotometric analysis of the resulting Th solution. The disodium salt of 2-(2-hydroxy-3,6-disulfo-1-naphthylazo)-benzenearsonic acid (Thorin) was used as the color forming reagent. The method gave satisfactory results in the range of 1 to 10 ppm Th. The concentration of B^{10} relative to B^{11} was determined by using the isotope exchange method of sample preparation and the mass spectrometer for isotopic analysis. The apparatus and tests of the chemical method are described. A rapid method is presented for the determination of B in Zr. The sample is decomposed by digesting it with H_2SO_4 at 220°C. By the exact control of the quantities of quinalizarin reagent, water, and H_2SO_4, the B in a suitable aliquot of the sample can be determined photometrically. A single step procedure which utilizes the filters of the A.R.L. intensity control stand was studied as a means of obtaining rapid analyses of B in Zr–U alloys. With this method extensive sample preparation was avoided, for the alloys were burned directly in a d.c. arc. A series of exposures were made through various filter combinations, then the B was densitometered and translated directly into concentration through the use of a calibration curve, which may be employed as an analytical curve. (auth)

9591 UCRL-1054
California. Univ., Berkeley. Radiation Lab.
CHEMISTRY DIVISION QUARTERLY REPORT [FOR] SEPTEMBER, OCTOBER, NOVEMBER 1950. Jan. 3, 1951. Decl. Mar. 22, 1957. 53p. Contract W-7405-eng-48. $9.30(ph OTS); $3.60(mf OTS).

ANALYTICAL PROCEDURES

9592 CF-56-7-106
Oak Ridge National Lab., Tenn.
NEUTRON ACTIVATION ANALYSIS. G. W. Leddicotte and S. A. Reynolds. July 20, 1956. 18p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The development of nuclear reactors has given the analyst a new analysis method which can be successfully applied to the determination of microgram and submicrogram quantities of many elements. Known as "activation analysis," this method is one in which an "activation" by some type of nuclear reaction is used to produce a radioactive isotope of the element to be determined. Since this radioisotope decays with its own characteristic radiations and half life, it is possible to make activation analysis a very specific analysis. Chemical separations of the radioisotope are employed whenever necessary and its radioactivity measured by some type of radiation counter. Of all the nuclear reactions, the neutron reactions have been used most often in activation analysis. The most useful neutron reaction, the n,γ reaction, involves the capture of a neutron by a stable nucleus to produce a radioisotope of the original atom. Since the Oak Ridge reactors are prolific sources of n,γ reactions, it has been possible to use activation analysis in many applications requiring an analytical method of great sensitivity and specificity. This paper briefly describes the program of neutron activation analysis service at Oak Ridge National Laboratory. The techniques used in this work and the limits of measurement and general applications of the method are also discussed. (auth)

9593 CF-56-8-122
[Oak Ridge National Lab., Tenn.]
SCIENTIFIC TECHNICIAN MANUAL—SAMPLING. G. E. Woodall. Aug. 7, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Basic information necessary for a technician to carry out a sampling program is given. Methods, quantities, containers, labelling, and forms to submit for analysis are discussed. (M.H.R.)

9594 FMPC-66
National Lead Co. of Ohio, Cincinnati.
THE DETERMINATION OF TRIBUTYL PHOSPHATE BY AN ACID SATURATION METHOD. E. A. Brown. July 7, 1952. Decl. Feb. 19, 1957. 14p. Contract AT(30-1)-1156. $0.25(OTS).

A procedure is described for the quantitative determination of tributyl phosphate (TBP) in kerosene. The method is based on the determination of the acid normality in the organic phase after treatment with excess nitric acid. Studies indicate that the quantity of TBP present determines the varying amount of available acid found in the organic phase. Actually, the acid distribution in the system is directly proportional to the TBP content for a given acid concentration. This fact is utilized in the analysis. Kerosenes from different sources were investigated, and this variable was found to have little effect on the reliability of the analyses. High results are obtained in the presence of uranium, thus necessitating its removal. The standard deviation of the method is 0.16 per cent. (auth)

9595 HW-25927
[Hanford Works, Richland, Wash.]
DETERMINATION OF AMERICIUM. H. R. Schmidt. Oct. 15, 1952. Decl. Feb. 28, 1957. 3p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

9596 NYO-2029
New Brunswick Lab., AEC., N. J.
PRELIMINARY REPORT ON ANALYSIS OF THORIUM METAL. Harold R. Mullin and Joseph J. Tregoning. [Aug. 13, 1952]. Decl. Feb. 28, 1957. 30p. $4.80(ph OTS); $2.70(mf OTS).

9597 NYO-5026
Mallinckrodt Chemical Works, St. Louis.
SHOTGUN TEST ON [URANIUM] DIOXIDE. C. R. Conard. Jan. 1, 1943. Decl. Feb. 26, 1957. 9p. $1.80(ph OTS); $1.80(mf OTS).

The sampling and ether extraction procedures for shotgun analysis of UO_2 are described. (T.R.H.)

9598 ORNL-153
Oak Ridge National Lab., Tenn.
A SOLVENT EXTRACTION METHOD FOR PLUTONIUM ANALYSIS. F. L. Moore and J. E. Hudgens, Jr. Sept. 2, 1948. Decl. Feb. 7, 1957. 18p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

A rapid and quantitative radiochemical method for the analysis of Pu is described. The method is based on the extraction of Pu^{+4} with TTA in C_6H_6 and a re-extraction into aqueous HF solution. Plutonium activity is recovered free from interferences, both radioactive and non-radioactive, and the method may be readily adapted to remote control. (auth)

9599 ORNL-1954
Oak Ridge National Lab., Tenn.
DETERMINATION OF CORROSION PRODUCTS AND ADDITIVES IN HOMOGENEOUS REACTOR FUEL. III. POLAROGRAPHIC DETERMINATION OF IRON(III). A. D. Horton and P. F. Thomason—H. P. Raaen, ed. Oct. 24,

1955. Decl. Mar. 1, 1957. 19p. Contract W-7405-eng-26. $0.25(OTS).

An ion-exchange — polarographic method was developed for the determination of Fe(III) in homogeneous reactor fuels. Copper, which interferes, is removed from the fuel by plating it onto a Cd coil. Iron is oxidized to Fe(III) by $KMnO_4$, and the Fe(III) is separated from interfering metal ions by ion exchange on a Dowex 1 resin column that is in the SO_4^{-2} form. The Fe(III) in the effluent is determined polarographically in 0.5 M sodium citrate solution as supporting electrolyte. A fairly well defined polarographic wave is obtained for the Fe(III) → Fe(II) reduction at a half-wave potential of approximately −0.15 v. vs. the S.C.E. The relative standard deviation of the data for 2 μg of Fe(III)/ml of solution in the polarographic cell was 6.5%; for 10 μg of Fe(III)/ml it was 0.6%. (auth)

9600 ORNL-2003
Oak Ridge National Lab., Tenn.
DETERMINATION OF TRACES OF URANIUM METAL BY DECOMPOSITION OF THE HYDRIDE. A. S. Meyer, Jr., B. L. McDowell, and J. C. White. Nov. 9, 1955. Decl. Mar. 2, 1957. 44p. Contract W-7405-eng-26. $0.40 (OTS).

Two methods were developed for the determination of U metal in UF_3 and mixtures of UF_3 with other fluoride compounds. A simplified method of determination which has a relatively high degree of precision (coefficient of variation 2%) is carried out by decomposing the hydride in an atmosphere of CO_2 and subsequently measuring the H_2 over an aqueous solution of KOH. The ignition of the hydride in an atmosphere of O_2 and volumetric measurement of the H_2O at reduced pressures provide a more sensitive method of determination. The coefficient of variation of the latter procedure is 7%. The yield of gas from the reaction of the hydride with NH_3 and gaseous HCl was found to be neither stoichiometric nor reproducible. (auth)

GRAPHITE

9601 BMI-T-26
Battelle Memorial Inst., Columbus, Ohio.
FABRICATION OF URANIA-BEARING GRAPHITE. H. Z. Schofield, J. A. Slyh, and L. D. Loch. May 15, 1950. Decl. Apr. 2, 1957. 17p. Contract W-7405-eng-92. $0.30 (OTS).

A method has been developed for the production of sound, uranium-bearing graphite bodies. In this process, uranium oxide is introduced into the graphite body in the form of discrete particles of controlled size, and the methods of fabrication are adjusted to permit random dispersion of these particles. In general, the process consists of two separate steps. In the first, nodules are prepared by pelletizing a mixture of uranium dioxide and shellac followed by crushing and sizing to the desired size. In the second, a mixture of artificial graphite powder and phenolic resin is prepared, the uranium nodules blended therein, and the resultant mixture formed into specimens and baked to set the binder C. To conserve U and to reduce the recoverable uranium loss to a minimum, forming of the specimens is done by compression molding in steel dies. The technique of fabrication, the equipment employed, and recoverable losses of U encountered are described in complete detail in this report. A flowsheet of the process and the specifications of the equipment are included in the Appendix. (auth)

9602 HW-21659
Hanford Works, Richland, Wash.
GRAPHITE SAMPLE BORING FROM PROCESS CHANNELS. Final Report on Production Test 105-389-P. J. B. Cole. July 10, 1951. Decl. Mar. 1, 1957. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

The equipment and techniques used in taking graphite samples from Hanford reactors are described. (D.E.B.)

9603 NAA-SR-153
North American Aviation, Inc., Downey, Calif.
THE MAGNETIC SUSCEPTIBILITY OF GRAPHITE AND THE EFFECT OF IRRADIATION ON ARTIFICIAL GRAPHITE. J. J. Donoghue and J. D. McClelland. Dec. 18, 1951. Decl. Mar. 28 1957. 18p. Contract AT-11-1-GEN-8. $0.30(OTS).

The change in the magnetic susceptibility of artificial graphite as a function of pile irradiation was determined using, for the first time, the invariant property of the total susceptibility. The latter is the sum of any three mutually perpendicular susceptibilities observed on a polycrystalline material. The invariance of the sum was established experimentally by numerous observations on artificial graphites. This property was employed to make precise observations of the decrease in the diamagnetic susceptibility of pile graphite with progressive exposure in a neutron flux. The results are in accord with the behavior predictable from the zone structure, and are consistent with the behavior observed for the electrical resisitivity, Hall coefficient, and magneto-resistivity. (auth)

9604 TID-10000
Brookhaven National Lab., Upton, N. Y.
THE ANNEALING OF THE GRAPHITE MODERATOR STRUCTURE IN THE BNL REACTOR. M. Fox and R. W. Powell. Jan. 1954. Decl. Mar. 12, 1957. 17p. $0.25 (OTS).

Data are presented from measurements of the magnitude of stored energy contained in reactor graphite, the effect of radiation damage on dimensional stability of graphite, the effects of annealing on graphite growth, and the effect of stored energy release procedures on measurements of physical growth and c-axis growth. Measurements were made on samples of graphite placed in the high flux region of the Brookhaven Reactor in the early days of its history and other samples mined from the graphite structure. Data are presented graphically. (C.H.)

9605 TID-10023
Brookhaven National Lab., Upton, N. Y.
REACTOR GRAPHITE PROPERTIES DETERMINED AT BROOKHAVEN NATIONAL LABORATORY. Justin Karp. May 1, 1950. Decl. Mar. 12, 1957. 28p. $0.30(OTS).

Refer also to abstract 9725.

LABORATORIES AND EQUIPMENT

9606 CF-56-8-133
Oak Ridge National Lab., Tenn.
A SMALL HIGH TEMPERATURE LABORATORY FURNACE. C. E. Curtis. Aug. 17, 1956. 19p. Contract [W-7405-eng 26]. $3.30(ph OTS); $2.40(mf OTS).

A small high-temperature furnace is described which is simple and comparatively cheap in construction and in which temperatures up to 2000°C or higher can be attained in 2 to 4 hours. Zirconia is employed as the refractory material; heat is supplied by oxyacetylene. Construction procedures, drawings and operational details are given. (auth)

9607 KAPL-537
Knolls Atomic Power Lab., Schenectady, N. Y.
THE BETA EXPERIMENT EQUIPMENT. Nuclear Engineering Course 1948 – 1949. D. R. Miller (Lecture Notes by A. M. Anderson). Mar. 31, 1949. Decl. Mar. 27, 1957.

9p. (Memo-NEC-12). $1.80(ph OTS); $1.80(mf OTS).

Apparatus and techniques used in irradiating U in Na at 450°C are described. Photographs of hot laboratory equipment are included. (D.E.B.)

9608 LRL-80

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

A HORIZONTAL PULSED LIQUID-LIQUID CONTACTOR. H. Schneider, W. J. Luke, and T. E. Hicks. Jan. 1954. Decl. Feb. 26, 1957. 32p. Contract AT(11-1)-74. $0.30 (OTS).

A new type of liquid-liquid contactor, a horizontal pulse extractor, has been tested on a laboratory scale with system succinic acid-butanol-water. Extraction performance is at least equivalent to a vertical pulsed column, and the unit is hydraulically stable. The HPE promises to have the mechanical simplicity of the VPC combined with the compactness of the horizontal mixer-settler. (auth)

RADIATION EFFECTS

9609 ANL-4939

Argonne National Lab., Lemont, Ill.

THE DECOMPOSITION OF LIGHT AND HEAVY WATER BORIC ACID SOLUTIONS BY CP-3' PILE RADIATIONS. Edwin J. Hart, William R. McDonell, and Sheffield Gordon. Dec. 3, 1952. Decl. Feb. 12, 1957. 22p. Contract W-31-109-eng-38. $0.25(OTS).

The behavior of light and heavy water solutions of boric acid toward pile radiations has been investigated as a function of boric acid concentration. A study has also been made of the effect of hydrogen, hydrogen peroxide, and potassium iodide on the radiation stability of boric acid solutions. Under the conditions of γ-ray and thermal neutron flux existing in the Argonne Heavy Water Reactor, solutions containing up to 0.02M boric acid in light water and 0.04M boric acid in heavy water may be irradiated without decomposition. At higher concentrations of boric acid, a decomposition of water occurs at a rate of 1.83mM hydrogen and hydrogen peroxide per minute per molar boric acid. This corresponds to a $G_{H_2} = 1.25$. The results are interpreted in terms of the availability of hydrogen and hydroxyl free radicals, which are formed primarily by the high γ-ray flux in the reactor, for recombining hydrogen and hydrogen peroxide, the decomposition products of the water produced mainly as a result of the $B^{10}(n,\alpha)Li^7$ reaction. (auth)

9610 ANL-5305

Argonne National Lab., Lemont, Ill.

STABILITY OF DIPHENYL UNDER FAST FLUX IRRADIATION IN A DYNAMIC SYSTEM. Sherman Greenberg. Oct. 1954. Decl. Feb. 15, 1957. 25p. Contract W-31-109-eng-38. $0.30(OTS).

Two runs, of 25 and 73.5 hours duration, respectively, were made with diphenyl heated to temperatures in the range of 300 to 450°F in a circulating loop in the Experimental Breeder Reactor to determine radiation stability and possibility of carbon deposition on heater surfaces. Estimates of the rate of radiation damage in diphenyl were made. These were based on the amount of diphenyl exposed to epithermal flux, the amount of radiological products produced in the system, and the calculated flux. These specific damage values, expressed as percentage of material in the flux zone damaged per 10^{18} nvt, range from 25 to 34 per cent. Some carbonaceous deposit was observed during the 74 hour run, but it could not definitely be attributed to irradiation. (auth)

9611 NAA-SR-1013

North American Aviation, Inc., Downey, Calif.

RADIATION EFFECTS QUARTERLY PROGRESS REPORT [FOR] JANUARY–MARCH 1954. F. E. Faris, ed. Aug. 15, 1954. Decl. Mar. 4, 1957. 48p. Contract AT-11-1-GEN-8. $7.80(ph OTS); $3.30(mf OTS).

The usual relation for a lattice heat conductor such as graphite predicts that the thermal conductivity and specific heat at low temperatures should have the same temperature dependence. The observed anomalous T^3 and T^2 dependence of graphite conductivity and specific heat was investigated, and an explanation of the effect is developed by postulating the presence of ungraphitized carbonaceous material forming a bond between graphite particles. Measurements of the thermal conductivity and resistivity of brominated graphite samples are reported for the temperature range of 10 to 300°K. The field dependence of the Hall coefficient and the magneto resistance of graphite were determined at fields of 5 to 15 kilogauss for a group of samples irradiated up to 12 Mwd/ct. The Hall potential was found to increase linearly with field whereas the magneto-resistivity coefficient changed its behavior with exposure. The nature of low-temperature annealing processes of radiation damage in graphite were studied by an analysis of several cyclotron-irradiated samples exposed at -170, 25, and 150°C. Measurements of the release of stored energy during quasi-isothermal annealing in a constant-temperature oven from 100 to 300°C have been continued. A study of lattice imperfections in Th was undertaken by an examination of the annealing of cold-worked and proton-irradiated Th. A prominent annealing state was found in both samples near room temperature, and another state in the irradiated sample occurred near 300°C. Additional measurements of the shear modulus of cold-worked and irradiated Cu at -170°C are reported. A revised analysis of the formation of colloidal alkali metal particles and excess alkali metal in the form of F centers during the thermal annealing of crystals is given with reference to experiments with KCl. (W.L.H.)

9612 ORNL-130

Oak Ridge National Lab., Tenn.

DECOMPOSITION OF WATER AND AQUEOUS SOLUTIONS UNDER PILE RADIATION. A. O. Allen, T. W. Davis, G. V. Elmore, J. A. Ghormley, B. M. Haines, and C. J. Hochanadel. Oct. 11, 1949. Decl. Mar. 7, 1957. 63p. Contract W-7405-eng-26. $0.45(OTS).

Purified water decomposes under Oak Ridge pile radiation to a slight extent only. The decomposition products, hydrogen and hydrogen peroxide, react with each other under radiation in aqueous solutions to re-form water (back reaction). When the concentration of dissolved products reaches a certain steady-state level, the back reaction rate becomes equal to the decomposition rate and reaction ceases. The steady-state level of hydrogen from purified water in the Oak Ridge pile at 25°C is about 20 micromoles per liter, corresponding to a hydrogen pressure over the water of only 20 mm of mercury. The presence of impurities markedly increases the steady-state level, and it drifts upwards with time because of slow contamination with impurities from the vessel walls. Some dissolved substances raise the steady-state level much more than others; in HCl and $CuSO_4$ solutions, hydrogen pressures of hundreds of atmospheres can be obtained, but H_3PO_4 and KOH solutions behave much like pure water. The steady-state level is markedly reduced by an increase in the temperature. Addition of a fraction of an atmosphere of hydrogen gas to purified water suppresses the decomposition almost completely; but addition of hydrogen peroxide or oxygen increases the decomposition, so that more hydrogen is ob-

tained from oxygenated water than from initially gas-free water. The results are explained in terms of the behavior of free radicals H and CH formed when radiation splits the water molecules. Radicals formed in high concentration in regions of high ionization density (tracks of proton recoils) frequently react with one another to give H_2 and H_2O_2 before they can diffuse out into the water. (auth)

SEPARATION PROCEDURES

9613 A-1046
Mallinckrodt Chemical Works, St. Louis.
BROWN OXIDE BY-PRODUCT RECOVERY. [PART] I. D. Copenhafer and W. A. Oppold. July 9, 1943. Decl. Feb. 12, 1957. 8p. $0.25(OTS).

Two possible methods are presented for obtaining U from carnotite sludge and their feasibility briefly investigated. The sludge contained the following constituents; 21.5% U_3O_8, 21.5% V_2O_3, 10.7% SO_4^{+2}, 7.84% Na, and H.1% Cl^- (W.L.H.)

9614 A-1048
Mallinckrodt Chemical Works, St. Louis.
BROWN OXIDE BY-PRODUCT RECOVERY-III. J. V. Opie and W. A. Oppold. Sept. 10, 1943. Decl. Feb. 12, 1957. 19p. $0.30(OTS).

The experiments conducted in development and testing of a process for recovery of U from carbonate residue press cake are described. These UO_2 process residues contain about 50% U and 80% recovery is sought. The steps evolved for the process consist of nitrate removal (by slurrying, caustic precipitation of U, and filtering and washing), then reduction and calcination followed by solvent extraction. (T.R.H.)

9615 A-1049
Mallinckrodt Chemical Works, St. Louis.
BROWN OXIDE BY-PRODUCT RECOVERY. [Part] IV. J. V. Opie and W. A. Oppold. Sept. 21, 1943. Decl. Feb. 12, 1957. 9p. $0.25(OTS).

Large-scale experiments on the use of refinery residues in the UO_2 process were carried out confirming the earlier small-scale results and showing the feasibility of the process for pilot scale operation. The hex liquor obtained from U_3O_8 produced in the process, extracts very nicely. Re-ignition of the U_3O_8 to oxidize Fe and burn off the C improves the extractability. The U_3O_8 produced contained from 0.068% to 0.245% V_2O_5 depending upon the methods used. U_3O_8/V_2O_5 was ~300 to 1000 in U_3O_8. The filtrates from the leaches in all cases contain about 0.34% of the U in the original material. The U_3O_8 from the leaching operation contained considerable C and reduced Fe. Re-ignition of the U_3O_8 in an oxidizing atmosphere oxidized the Fe and burned off the C, making the final extraction much better. Variations in method of treatment and results obtained are tabulated. (L.D.)

9616 A-1050
Mallinckrodt Chemical Works, St. Louis.
BROWN OXIDE BY-PRODUCT RECOVERY. [Part] V. J. W. Barnes, D. R. Copenhafer, T. C. Furnas, J. V. Opie, and G. C. Reid. Oct. 28, 1943. Decl. Feb. 12, 1957. 14p. $3.30(ph OTS); $2.40(mf OTS).

Results are presented from an investigation to determine the feasibility of a proposed process for the recovery of uranium in residues from the brown oxide process. The proposed process was to consist essentially of extracting uranium from a water solution by bubbling ether through it. The relatively pure uranium could then be recovered from the ether by scrubbing with water, by evaporation, or by other suitable means. (C.H.)

9617 CF-52-3-215
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR CHEMICAL PROCESSING. Quarterly Report for the Period Ending March 31, 1952. D. E. Ferguson. Mar. 25, 1952. Decl. Feb. 20, 1957. 25p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70 (mf OTS).

The development of homogeneous reactor chemical processing has been based on a Pu producer using a UO_2SO_4 heavy water solution as fuel. The general approach has been to study application of tributyl phosphate solvent extraction to this separation problem, and to use this process as a yardstick by which to evaluate attractive alternate methods of separation. The work has now progressed to the point at which a reliable chemical flowsheet has been obtained; however, this flowsheet does not necessarily represent the optimum method of chemical processing. For instance, the inherent advantage of starting with U in solution is lost because of the necessity of heavy water recovery, and the long cooling period for Np decay necessitates large storage facilities and inventories of valuable materials. (auth)

9618 CF-53-6-173
Oak Ridge National Lab., Tenn.
ECONOMICS OF CHEMICAL PROCESSING FOR A URANIUM-233 BREEDER. Don E. Ferguson and John P. Sanders. June 18, 1953. Decl. Mar. 12, 1957. 25p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

9619 CF-55-4-72(Del.)
Oak Ridge National Lab., Tenn.
25 PROCESS ASSISTANCE FOR PERIOD OF MARCH 21, 1955 TO APRIL 1, 1955. F. L. Culler. Apr. 11, 1955. Decl. with deletions Feb. 14, 1957. 7p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Use of Thorex-type high temperature feed digestion at high HNO_3 concentrations will prevent subsequent organic emulsion formation in the 25 extraction column, even when the dissolver solution is known to contain emulsion-producing impurities. Gross D.F.'s for three spiked-feed small pulse column runs were at least 3×10^4 β and 5×10^3 γ across both columns. Trickle dissolver tests using 2S Al sheet in the 5-in. D pipe unit yielded a 5 ft packed height for a processing rate of 200 kg Al/day, equivalent to ~20 kg U/day when dissolving SR tubular elements. The packed height dropped to just under 3 ft when the dissolving rate was cut in half. Thorium metal quality studies indicate that metal samples prepared by arc-melting sintered compacts are within target specifications except for the iron content. The Abbie "Dispersall" mixer has been tested briefly for the dry-$ThCl_4$ contacting with NaHg and probably will prove satisfactory for making 0.5 lb batches of Th as quasi amalgam. (auth)

9620 CF-56-9-83
Oak Ridge National Lab., Tenn.
SEPARATION AND DETERMINATION OF NEPTUNIUM BY LIQUID-LIQUID EXTRACTION. Fletcher L. Moore. Sept. 25, 1956. 12p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A rapid and quantitative radiochemical method for the determination of Np^{237} or Np^{239} tracer is described. The method is based on the liquid-liquid extraction of Np(IV) into 0.5M 2-thenoyltrifluoroacetone–xylene. Neptunium is separated free from interferences, both radioactive and non-radioactive, and the method may be adapted readily to remote control. The technique has also been found to be very effective in the purification of Np tracer. (auth)

9621 CF-56-11-24
Oak Ridge National Lab., Tenn.
SUGGESTED PROCEDURE FOR OPERATION OF THE DIS-

SOLVER SYSTEM. Herman O. Weeren. Nov. 5, 1956. 7p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The operating procedure of the dissolver system in the HRT chemical plant is outlined. (B.J.H.)

9622 CN-3274

Hanford Engineer Works, Richland, Wash.

SUMMARY OF RESULTS OF INVESTIGATIONS ON THE INFLUENCE OF HYDRAZINE IN THE BISMUTH PHOSPHATE EXTRACTION STEP. D. R. Miller. June 11, 1945. Decl. Feb. 16, 1957. 5p. (SE-PC-45). $1.80(ph OTS); $1.80(mf OTS).

This report reissued under auspices of Contract W-7405-eng-37.

Results of an investigation on the influence of hydrazine in the bismuth phosphate extraction step are presented. (W.L.H.)

9623 FMPC-427

National Lead Co. of Ohio, Cincinnati.

EXTRACTION COLUMN RAFFINATE CONTROL. B. G. Ryle, E. J. Beer, and J. F. Blum. May 1954. Decl. Feb. 19, 1957. 21p. Contract AT(30-1)-1156. $0.30(OTS).

An Ohmart density measurement system has been evaluated in the FMPC pilot plant as an instrumental method of preventing U losses in the extraction column raffinate stream. It has been established that the system is capable of detecting the presence of low U concentrations in a multicomponent, two-phase system as produced in the tributyl phosphate-nitric acid-uranium extraction process. (auth)

9624 HW-32175

Hanford Atomic Products Operation, Richland, Wash.

A STUDY OF CERTAIN PROPERTIES OF RUTHENIUM COMPOUNDS FOUND IN THE REDOX PROCESS. W. E. Gill and D. E. Wisehart. June 23, 1954. Decl. Mar. 13, 1957. 8p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

In a study of the penetration by Ru compounds, silica gel was found to have an efficiency of 99.9+% for collection of gaseous RuO_4 and droplets and particles from the Ru off-gas line. A temperature of 90 to 95°C did not in itself cause decomposition of RuO_4 when it was distilled from an aqueous solution. (F.S.)

9625 IDO-14344(Del.)

Phillips Petroleum Co. Atomic Energy Div., Idaho Falls, Idaho.

PILOT PLANT DEVELOPMENT OF A PROCESS FOR SEPARATING BARIUM-140 FROM MTR FUEL. B. M. Legler, W. L. Chase, M. D. Martin, D. M. Paige, and F. K. Wrigley. Sept. 1, 1955. Decl. with deletions Feb. 11, 1957. 169p. Contract AT(10-1)-205. $25.80(ph OTS); $7.80(mf OTS).

Extensive full-scale pilot plant studies, using unirradiated materials, of a precipitation process designed to separate 30,000 to 50,000 curie quantities of high purity Ba^{140} from a twenty-day-irradiated, two-day-cooled MTR fuel assembly were conducted. Centrifugation was employed to effect liquid-solid separations. This process consists generally of fuel dissolution in five molar NaOH followed by centrifugation of dissolver slurry to remove Al; dissolution of caustic residue in dilute HNO_3 followed by centrifugation to remove Ba-containing solution from silica and insoluble particulates; precipitation of $Ba(NO_3)_2$ by solution concentration and concentrated HNO_3 precipitant addition followed by centrifugation to remove U and fission products; double precipitation of barium chromate from aqueous crude product solution followed by centrifugation to remove Sr; metathesis to and final precipitation of $Ba(NO_3)_2$ followed by centrifugation to remove remaining product impurities; and product drying. Pilot plant operation employing inactive Ba and simulated fission product spike to dissolver cake yielded Ba recoveries of 70 to 90% and high product purity. The process indicates good reproducibility and operability with an over-all operating time of approximately 16 hours. Waste U bearing solution is suitable for recovery of U in existing Idaho Chemical Processing Plant facilities. Unrecoverable uranium amounts to approximately 2% of that charged. (auth)

9626 KLX-1201

Kellex Corp., New York.

PROGRESS REPORT FOR THE PERIOD MARCH 1 – 31, 1950. Job 23. April 7, 1950. Decl. Mar. 5, 1957. 23p. Contract AT(30-1)-848. $4.80(ph OTS); $2.70(mf OTS).

The testing of organic solvents for solvent extraction of U is reported. Several anion exchange resins are evaluated for their adsorption of U. The extraction of Ra from U refinery operations residues using 100% H_2SO_4 is described. (T.R.H.)

9627 KLX-1203

Kellex Corp., New York.

PROGRESS REPORT FOR THE PERIOD MAY 1 – 31, 1950. Job 23. June 5, 1950. Decl. Mar. 2, 1957. 36p. Contract AT(30-1)848. $6.30(ph OTS); $3.00(mf OTS).

Studies are continued in the development of solvents and diluents for U solvent extraction. Other research in the factors affecting adsorption of U on anion exchange resins is reported. Methods for recovery of Ra from U ore acid leach residues are investigated, and experiments in purification of Ra are described. (T.R.H.)

9628 KLX-1209

Kellex Corp., New York.

PROGRESS REPORT FOR NOVEMBER 1950. Job 23. Jan. 3, 1951. Decl. Feb. 27, 1957. 43p. Contract AT(30-1)-848. $7.80(ph OTS); $3.30(mf OTS).

Progress is reported on the solvent leaching of high grade ore and MgX, with emphasis on larger-scale processing of pitchblende and purification by liquid-liquid extraction of the resultant pregnant leach liquor; liquid-waste treatment by ion-exchange methods; recovery of U from solid wastes; evaluation of selected solvents as substitutes for diethyl ether in U-extraction processes; and the applicability of solvent extraction, distillation, and thermal decomposition for recovering acid values in raffinates. (D.E.B.)

9629 KLX-1220

Vitro Corp. of America, New York.

LIQUID WASTE TREATMENT (23C). Summation Report for the Period February 1950 through April 1951. Job 23. Nov. 21, 1951. Decl. Feb. 28, 1957. 14p. Contract AT(30-1)848. $3.30(ph OTS); $2.40(mf OTS).

On the basis of results given, it appears that U can be removed economically from MCW main sewer liquor, using Dowex-2 resin. It is possible to recover U from MCW C-3 discard liquor by adsorption on aminized cotton and from MCW acid raffinate and MCW acid sewer liquor by adsorption on phosphorylated cotton cloth. The recovery of U from Peterson discard liquor by ion exchange appears impractical because of formation of precipitates within the column. It does not appear economically feasible to recover U from sump recovery liquor. Two composites representing the proposed "new refinery" waste streams were investigated. With one composite, representing a plant operating with C-3 process and no acid sewer, adsorption by ion exchange is questionable, since a precipitate which forms slowly on mixing the components would probably interfere with column operation. With the other composite representing the combined sewers from a plant using the

"Slurex" process, the volume of waste treated per cu ft of resin and the column loading are low, which would make the cost per pound of U_3O_8 recovered unduly high. From the health physics point of view, adequate decontamination of the liquor is obtained and for this reason operation of the process may be warranted. (D.E.B.)

9630 KLX-1221

Vitro Corp. of America, New York.

GANGUE REDIGESTION (23H). Summation Report [for] Period December 1950 through April 1951. Job 23. Jan. 28, 1952. Decl. Feb. 28, 1957. 27p. Contract AT(30-1)-848. $0.35(OTS).

Methods for recovering U from Pb gangue were investigated. A process which comprises treating gangue cake with H_2SO_4 and adsorbing U from the clear leach liquor by ion exchange was developed. This process was demonstrated on laboratory and pilot scales. (J.E.D.)

9631 KLX-1223

Vitro Corp. of America, New York.

RECOVERY OF RADIUM FROM K-65 RESIDUE (23-E). Complete Report [for] March 1950 through September 1950; May 1951 through November 1951. Job 23. Feb. 14, 1952. Decl. Feb. 28, 1957. 53p. Contract AT(30-1)-848. $0.45 (OTS).

Methods for recovering Ra from solid residues from U refining operations have been investigated. A process using Versene was developed and demonstrated on a laboratory scale. In this process the residue is leached with Versene solution and Ra is adsorbed from the leach liquors and purified by the use of cation exchange resin. (C.H.)

9632 KLX-1224(Del.)

Vitro Corp. of America, New York.

URANIUM RECOVERY FROM SCRAP MATERIAL (23-J). Project Summation [Covering the period] January 1951–January 1952. Job 23. June 16, 1952. Decl. with deletions Feb. 26, 1957. 63p. Contract AT(30-1)-848. $10.80(ph OTS); 3.90(mf OTS).

Methods have been investigated for the recovery of U from various C-Scrap by-product materials resulting from the production of U metal by bomb reduction of UF_4 with Mg. A process developed and demonstrated on a laboratory scale consists of roasting the materials, grinding to 200 mesh, digesting with hot HCl plus added chlorate at ph < 0.5, treating the clarified digest liquor with NH_3 by continuous precipitation procedure and roasting the washed precipitate to form black oxide. The black oxide produced by this process contains over 50% U_3O_8 and less than 0.1% F, and is free of Cl. The by-product insolubles contain less than 0.05% U_3O_8 and the waste liquor is below the U and F tolerance levels. (auth)

9633 KLX-1704

Vitro Corp. of America, New York.

HOMOGENEOUS REACTOR FUEL REPROCESSING. Quarterly Progress Report No. 1 for the Period July 1–September 30, 1953. Job 87. Oct. 19, 1953. Decl. Feb. 27, 1957. 16p. For Oak Ridge National Lab. Contract W-7405-eng-26, Subcontract 535. $3.30(ph OTS); $2.40 (mf OTS).

In a study of solubilization conditions of rare-earth fluorides in 0.02M UO_2SO_4 at 100°C, it was found necessary to have about 0.3 g of residual fluoride per liter in solution to maintain the rare earth-fluorides insoluble. Tests are reported for rare-earth absorption by CaF_2 at 100°C, fused alundum pellets impregnated with 1% CaF_2, garnet minerals, Ca titano-silicate mineral, and a natural Ca Mg silicate (diopside). Treatment of effluent from a CaF_2-packed absorption column with stoichiometric Th ion plus 10% excess reduced the fluoride ion to 90 ppm with negligible U removal; anhydrous $CaSO_4$ in contact with a similar effluent reduced the Ca content from 0.6 to 0.042 g/l, also with negligible U removal. (L.M.T.)

9634 MCW-4

Mallinckrodt Chemical Works, St. Louis.

A STUDY OF ETHER EXTRACTION OF URANYL NITRATE HEXAHYDRATE LIQUORS WITH SPECIAL REFERENCE TO THE FORMATION OF EMULSIONS. Harold Yeager. Apr. 1, 1946. Decl. Feb. 26, 1957. 23p. Contract W-7405-eng-1. $0.35(OTS).

A study of the characteristics of emulsion formation of ether and impure uranyl nitrate hexahydrate being mixed by a centrifugal pump showed that high ratios of water phase to ether phase being mixed caused emulsions and that low ratios prevented or broke emulsions. The extractors were altered in such a way that the ether and water phases were separated in the extractor while ether was being recirculated from a side outlet in the extractor through two pumps, with impure molten uranyl nitrate being added to the inlet side of the second pump. No further trouble with emulsions was encountered. (B.J.H.)

9635 MCW-14

Mallinckrodt Chemical Works, St. Louis.

PURIFICATION OF PHOSPHATE RESIDUE FROM PREVIOUS DIGESTIONS OF URANIUM OXIDES FOR ETHER EXTRACTION. Wayne H. Keller. Apr. 1, 1946. Decl. Feb. 26, 1957. 17p. Contract W-7405-eng-1. $0.30(OTS).

Filter cake, known as 306 residues, consisting chiefly of uranyl phosphate and Na uranate, is accumulated in large quantities in the production of UO_2 from UO_3. Detailed studies on the HNO_3 solution of these residues, filtration of the digest liquors, concentration by evaporation, salting-out agents, and ether extraction are reported. A process based on the results of these studies is recommended for U recovery from these residues. (D.E.B.)

9636 MCW-254

Mallinckrodt Chemical Works, St. Louis.

PRODUCTION OF URANIUM METAL FROM UNH FROM HANFORD SLUGS DECONTAMINATED BY THE REDOX PROCESS. W. H. Keller. Apr. 11, 1950. Decl. Feb. 27, 1957. 10p. $0.25(OTS).

Denitration to UO_3 of decontaminated $UO_2(NO_3)_2$ liquor, reduction of UO_3 to UO_2, fluorination of UO_2 to UF_4, reduction of UF_4 to metal, and casting of the metal in vacuum are described. (W.L.H.)

9637 NAA-SR-213

North American Aviation, Inc., Downey, Calif.

THE USE OF BROMINE TRIFLUORIDE IN THE RECOVERY OF URANIUM FROM URANIUM IMPREGNATED GRAPHITE FUEL MATERIALS. F. D. Rosen. Dec. 29, 1952. Decl. Feb. 27, 1957. 34p. $0.30(OTS).

Removal of 99.8% of the U as UF_6 from graphite impregnated with uranium oxide or carbide has been accomplished by treating the material with at least 0.5 mole of BrF_3 per mole of graphite. A preliminary investigation of the reaction of BrF_3 with graphite indicated that the following reactions may occur, depending on the experimental conditions: $BrF_3 + C \rightleftharpoons C_nBrF_3$ ($n \geq 1$), $BrF_3 + C \longrightarrow (CF)_x + Br_2$, $BrF_3 + C \xrightarrow{UC_2} CF_4 + Br_2$. Graphite impregnated with UC_2 has been found to be considerably more reactive toward BrF_3 than the oxide-impregnated materials, but preliminary treatment with liquid Br_2 ensured a smooth reaction on subsequent addition of BrF_3. Safety requirements for handling the BrF_3-graphite system are reviewed, with special reference to the dry graphite residue left after BrF_3 treatment. (auth)

9638 NAA-SR-214

North American Aviation, Inc., Downey, Calif.

THE SEPARATION OF URANIUM FROM ZR–U FUEL BY

FLUORINATION METHODS. W. Gardner. Dec. 29, 1952. Decl. Mar. 2, 1957. 18p. Contract AT-11-1-GEN-8. $0.25(OTS).

Two fluorination processes are reported for the separation of U from U–Zr fuel material (96 wt%). The first is a two-step fluorination in which the fuel alloy is dissolved in very concentrated aqueous HF, the product fluorides are dried, and then the UF_4 is converted to UF_6 with liquid BrF_3. The second method is a direct reaction of solid U–Zr alloy at 800°C with elemental F to produce ZrF_4 and volatile UF_6. A preliminary investigation of corrosion problems is reported for both processes. (auth)

9639 NAA-SR-1374

Atomics International Div., North American Aviation, Inc., Canoga Park, Calif.

INVESTIGATION OF PYROCHEMICAL PROCESSING OF Th–U FUEL. Semi-Annual Report [for] September 1954–February 1955. F. J. Keneshea, Jr., A. G. Buyers, and E. E. Motta. Jan. 15, 1957. Decl. Mar. 14, 1957. 18p. Contract AT-11-1-GEN-8. $0.30(OTS).

Separation of Ce, rare earths, Cs, Sr, and Te from irradiated Th–U was accomplished by volatilization from a Ta crucible, leaving Pa and Ru. With a TiC crucible the same thing occurred. In a ThO_2 crucible 50% removal of Ce and rare earths occurred, and in ZrO_2 70% separation occurred. In a five-pass arc-zone-melting run Cs and Sr were completely volatilized leaving Ce and the rare earths. (T.R.H.)

9640 NAA-SR-Memo-125

North American Aviation, Inc., [Downey, Calif.]

HTR CHEMICAL PROCESSING—BISMUTH TRICHLORIDE AS A DECONTAMINANT FOR BISMUTH. D. Cubicciotti. Dec. 19, 1951. Decl. Feb. 28, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

The separation and removal of fission products from liquid Bi by treatment with fused $BiCl_3$ was demonstrated. A one-step process for the fission product decontamination of Bi coolant is suggested. (C.W.H.)

9641 NYO-1176

Catalytic Construction Co., Philadelphia.

MULTICOMPONENT EXTRACTION CALCULATION PROCEDURE. LONG RANGE SURVEY PROGRAM. Job No. 3900. F. W. Winn. Jan. 1, 1953. Decl. Mar. 1, 1957. 27p. Contract AT(30-1)-1311. $4.80(ph OTS); $2.70(mf OTS).

Detailed descriptions are given of the calculation procedure for use in evaluating proposed extraction systems prior to pilot-plant demonstrations. The procedure is applicable to extraction, stripping, and compound columns and is also used to compute the composition of the two phases on all trays in the column. The procedure is illustrated by application to a Th purification problem in a compound column. (J.A.G.)

9642 NYO-1305

Mallinckrodt Chemical Works, St. Louis.

MCW [MALLINCKRODT SF] ACCOUNTABILITY PROGRAM. Progress Report for Apr. 15, 1950 to June 1, 1950. C. H. Walden. June 7, 1950. Decl. Feb. 28, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

The present practice in checkweighing UO_3 and UO_2 is described briefly, and operation of the discard-liquor sampler is summarized. Accountability problems related to disposal of "hot scrap" from sumps, etc., and combustible and non-combustible trash are discussed. The status of installation of a sampling station in the U metal plant is stated. (G.Y.)

9643 NYO-1337

Mallinckrodt Chemical Works, St. Louis.

JET-MIXER COLUMN FOR THE: (1) EXTRACTION OF URANIUM FROM VARIOUS REFINERY FEEDS, AND (2) EXTRACTION OF NITRIC ACID FROM RAFFINATE. Period Covered Feb. 11, 1950 to June 25, 1951. J. H. Yeager and K. S. Eckberg. May 1, 1952. Decl. Mar. 2, 1957. 29p. Contract W-14-108-eng-8. $0.35(OTS).

This investigation involved the development and evaluation of a pilot plant jet-mixer column for the extraction of U from both pitchblende and unfiltered GFA type feeds with acid ether, and for the extraction of HNO_3 from raffinate with neutral ether. (auth)

9644 NYO-1347

Mallinckrodt Chemical Works, St. Louis.

USE OF SURFACE ACTIVE AGENTS TO PREVENT EMULSIONS IN SLUREX EXTRACTIONS. R. Piper and A. E. Ruehle. Aug. 11, 1952. Decl. Feb. 28, 1957. 15p. Contract W-14-108-eng-8. $0.25(OTS).

Preliminary investigations indicate that several surface-active agents are effective in preventing the formation of emulsions in the extraction of uranium from pitchblende slurry with TBP diluted with kerosene or with CCl_4. About 50 surface-active agents have been tested with both diluents in cylinder shakeouts and those found effective run in a Scheibel column. The use of surface-active agents has given satisfactory operation in Scheibel column runs both with kerosene and with CCl_4 under conditions which would otherwise produce emulsions. (auth)

9645 NYO-1464

Harshaw Chemical Co., Cleveland.

PROGRESS REPORT FOR NOVEMBER 1951. J. T. Pinkston and Karl S. Willson. Dec. 1, 1951. Decl. Feb. 27, 1957. 7p. Contract W-7405-eng-276. $1.80(ph OTS); $1.80(mf OTS).

Brief statements are presented covering the completion and capacity of the plants for the extraction of uranyl nitrate using a TBP-kerosene solvent. Flow rates and possible means of minimizing solvent loss are studied. (J.E.D.)

9646 NYO-1467

Harshaw Chemical Co., Cleveland.

CLOSING RESEARCH REPORT, DECEMBER 31, 1951. J. T. Pinkston and Karl S. Willson. Feb. 15, 1952. Decl. Feb. 27, 1957. 13p. Contract W-7405-eng-276. $3.30(ph OTS); $2.40(mf OTS).

The work done on the development and plant scale installation of a process for U refining using mixer-settler equipment and a TBP-kerosene mixture as a solvent for the countercurrent extraction of uranyl nitrate from raw slurries is summarized. (J.E.D.)

9647 NYO-5108

[Mallinckrodt Chemical Works, St. Louis.]

PITCHBLENDE PROCESSING FILTRATION DATA. Engineering B Data Progress Report I for April 3–14, 1945. Earl Miller. Apr. 21, 1945. Decl. Mar. 8, 1957. 13p. $3.30(ph OTS); $2.40(mf OTS).

Preliminary filtration data are presented and tabulated on relatively large batches so that the type and size of filtration equipment needed for plant-scale operations of the pitchblende ore process can be decided upon. Subsequent experiments will serve to provide needed engineering data. (J.E.D.)

9648 NYO-5113

Mallinckrodt Chemical Works, St. Louis.

A STUDY OF CONTINUOUS EXTRACTION METHODS. Thomas C. Furnas, Jr. May 12, 1945. Decl. Mar. 7, 1957. 181p. Contract W-7405-eng-1. $28.80(ph OTS); $8.40(mf OTS).

A number of experiments on the first U ether extraction step are described and the concentration distribution equilibrium curves given. The most desirable extraction conditions are obtained by using a feed liquor which con-

tains both an excess of nitric acid and a salting out agent such as calcium nitrate and a solvent containing nitric acid. Under these conditions the equilibrium curve turns sharply upward from the origin and maintains a fairly steep slope which considerably reduces the number of theoretical plates required to attain the degree of "hex" extraction desired. The use of sodium nitrate as a salting-out agent was not satisfactory and the slope of the distribution curve in this instance was not improved. Based on these results, laboratory experiments using towers were conducted for carrying out extraction, washing, and stripping operations in a continuous manner. Experimental results are reported for the extraction of "hex" = $MO_2(NO_3)_2\ 6H_2O$ from water liquors by ether. The packed tower seemed to be most satisfactory. The spray type tower worked very well for washing the ether extract and also did a very thorough job of stripping the pure "hex" from the washed ether solution. These towers are described briefly in the experiments and are shown in diagrams. Pilot plant experiments are reported, results are given, and the pilot plant set up is illustrated. (L.D.)

9649 NYO-5135
Mallinckrodt Chemical Works, St. Louis.
[PITCHBLENDE PROCESSING DEVELOPMENT.] PRELIMINARY TESTS ON OLIVER FILTER, BATCHES S-104 INCLUSIVE. C. F. Ritchie. Oct. 15, 1945. Decl. Mar. 8, 1957. 20p. $4.80(ph OTS); $2.70(mf OTS).

The Oliver drum filter was tested quite extensively on two types of precipitate resulting from the digestion of pitchblende ore and its subsequent treatments. The solids resulting from digestions wherein the acid was added to the ore slurry were of a bulky gelatinous nature. When this precipitate was filtered at lower temperature (140 to 160°F) good cakes were formed, but these were so very soft that application of wash water was virtually impossible. A second type of precipitate was produced by adding the ore to the acid. This formed a much more dense, nongelatinous filter cake which had much less cracking tendency and accepted water better. Previously, the Feinc string discharge filter had been ruled out for handling the soft precipitate, but with the discovery of the technique for preparing a dense precipitate, it was believed that the Feinc could be made to operation. (C.J.E.)

9650 NYO-5156
Mallinckrodt Chemical Works, St. Louis.
DEVELOPMENT OF A CONTINUOUS ETHER EXTRACTION PROCESS FOR PURIFICATION OF URANIUM DERIVED FROM AAA [PITCHBLENDE] ORE. J. V. Opie, W. H. Keller, and J. H. Yeager. Jan. 15, 1946. Decl. Mar. 8, 1957. 95p. Contract W-14-108-eng-8. $16.80(ph OTS); $5.70(mf OTS).

The development of continuous process equipment and investigation of chemical and physical variables operating conditions, and purity of products are described for the continuous extraction of U from AAA pitchblende ores. (W.L.H.)

9651 NYO-5168
Mallinckrodt Chemical Works, St. Louis.
A REPORT ON THE REDUCTION OF MoO_3 IN THE ETHER EXTRACT FROM AAA [PITCHBLENDE] ORE ACID FEED SOLUTIONS. James Miller. Feb. 22, 1946. Decl. Mar. 7, 1957. 11p. Contract W-14-108-eng-8. $3.30(ph OTS); $2.40(mf OTS).

In preparation pitchblende ore acid feed solutions for the first continuous ether extraction, high-temperature boil-down, long reflux period, and use of $Fe(NO_3)_3$ as a salting agent all reduce the MoO_3 extraction into ether. It would be possible to design a one-step process for ether extracting a pure $UO_2(NO_3)_2$ from the acid feed; however, its control would be difficult, since the pitchblende ore differ in composition. (W.L.H.)

9652 NYO-5196
Mallinckrodt Chemical Works, St. Louis.
A REPORT ON AN OUTLINE OF THE FUNDAMENTAL CHEMISTRY OF COLUMN FEED LIQUORS. Carl W. Kuhlman, Jr. Sept. 9, 1946. Decl. Mar. 7, 1957. 57p. Contract W-14-108-eng-8. $9.30(ph OTS); $3.60(mf OTS).

The work presented is designed as an outline of the fundamental chemistry of the system of ions present in the column feed liquors. Under certain conditions accidentally encountered under present plant procedures, e.g., too low acidities, precipitates form in these liquors making the extraction operation very difficult and converting some U into a nonextractible form. It is these precipitates that are chiefly treated in this paper. Owing to the great complexity of the column feed liquors, work was done on a series of simpler ion combinations to obtain data to serve as a guide in the treatment of the more complex systems encountered in the plant. These experiments cover the following general topics: composition of precipitates formed; solubility of precipitates formed; rate of precipitation; effects of complex forming ions on the solubility of these precipitates; and reversibility of the solubility relationships existing. (auth)

9653 ORNL-20
Oak Ridge National Lab., Tenn.
COMPARISON OF BATCH AND CONTINUOUS 23 PROCESSES. G. H. Hanson. March 17, 1948. Decl. Mar. 7, 1957. 39p. $15.30(ph OTS); $5.40(mf OTS).

The separation of U and Th by solvent extraction has been studied at X-10 and other sites for several years. In connection with the recovery and decontamination of U^{233} produced by the irradiation of Th in the proposed Research Pile and other piles, both batch and continuous solvent extraction processes have been proposed. For a plant capacity of fifteen kg of Th per day, a continuous column process is more favorable than one using intermittent batch extractions. This recommendation is contingent upon the economic solution of the crude problem and the demonstration of trouble-free operation with packed columns for long periods of time. The estimated investment and operating costs for each process are essentially the same. (D.E.B.)

9654 ORNL-122
Oak Ridge National Lab., Tenn.
LABORATORY PURIFICATION AND CONCENTRATION OF U^{233}. D. C. Overholt Aug. 23, 1949. Decl. Mar. 6, 1957. 20p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

A laboratory final concentration and purification process for U^{233} was developed to follow the recovery and decontamination of this material from bombarded Th. The procedure employs a batch-wise solvent extraction cycle with diisopropyl ether, evaporation of the product to dryness, and ignition to the oxide. The apparatus used for the concentration of U^{233} consisted of a jacketed extraction vessel with a total capacity of about 1500 ml. This vessel was equipped with a 10 × 1 in. interface stirrer and a coarse sintered glass filter which was sealed into the bottom. After agitation both organic and aqueous phases were drawn through the filter disc into a 1500 ml settling vessel. As the phases separated the aqueous was drawn off to a feed (1000 ml) hold-up vessel and the organic phase transferred to a stripping vessel. The aqueous phase was then returned to the extraction vessel and the cycle repeated. The stripping vessel consisted of a four-liter conical shaped vessel in which the two phases were agitated by a 10 × 1 in. interface stirrer, from which the strip product was drawn off for final volume reduction and the organic waste stored for

steam distillation and reuse. The extraction loss of U^{233} in this process is about 0.03%. Spectrographic analysis of a product solution indicated less than 0.2 mg Al per mg of U^{233} and only traces (100 to 1000 ppm) of Th, Ca, Mg, and Si. Isotopic analysis using a pulse analyzer showed 0 to 3% contaminating α activity. The β activity in the product appeared to be no greater than the G-M countable activity normally associated with the pure U^{233}. Alternate concentration methods involving precipitation with H_2O_2 and with tetramethyl ammonium hydroxide gave adequate separation from ionic contaminants but little or no decontamination from fission products. (auth)

9655 ORNL-260(Del.)
Oak Ridge National Lab., Tenn.
TRIBUTYL PHOSPHATE SOLVENT EXTRACTION OF URANIUM FROM METAL WASTE – LABORATORY INVESTIGATION. D. E. Ferguson and T. C. Runion. Oct. 7, 1949. Decl. with deletions Feb. 8, 1957. 47p. Contract W-7405-Eng-26. $7.80(ph OTS); $3.30(mf OTS).

A solvent extraction process employing TBP, diluted with hexane, as the extracting solvent and HNO_3 as the salting agent has been developed for the recovery of U from metal waste solutions. (C.W.H.)

9656 ORNL-302
Oak Ridge National Lab., Tenn.
MANGANESE DIOXIDE AS A FILTER AID IN THE "25" PROCESS. R. E. Blanco. July 27, 1949. Decl. Mar. 4, 1957. 17p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

An evaluation of MnO_2 as a filter aid for clarifying 25 Process metal solutions (U – Al fuel rods in HNO_3) is presented. (auth)

9657 ORNL-543
Oak Ridge National Lab., Tenn.
TBP PROCESS PILOT PLANT DESIGN REPORT. R. P. Milford. Jan. 27, 1950. Decl. Mar. 1, 1957. 38p. Contract W-7405-eng-26. $7.80(ph OTS); $3.30(mf OTS).

A set of stainless steel columns and accompanying process equipment, including a product evaporator, was installed in Bldg. 706-HB to demonstrate the TBP Process for recovering uranium metal waste. Tributyl phosphate, diluted with Varsol, is used as an extractant in one cycle of continuous liquid-liquid countercurrent extraction, scrubbing, and stripping. (auth)

9658 ORNL-564(Del.)
Oak Ridge National Lab., Tenn.
ORIENTATION MANUAL FOR 23 PILOT PLANT. D. E. Ferguson and R. J. Klotzbach. Feb. 15, 1950. Decl. with deletions Mar. 7, 1957. 37p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

The status of process chemistry for the recovery of 23 from pile irradiated thorium is summarized and the 23 Pilot Plant which will be used at Oak Ridge National Laboratory for further process development is described. (auth)

9659 ORNL-620
Oak Ridge National Lab., Tenn.
LABORATORY DEVELOPMENT OF AN ION EXCHANGE RALA PROCESS. R. E. Blanco. June 2, 1950. Decl. Feb. 28, 1957. 55p. Contract W-7405-eng-26. $9.30(ph OTS); $3.60(mf OTS).

This report presents the laboratory development of an ion exchange process for the purification of radioactive barium. (auth)

9660 ORNL-801
Oak Ridge National Lab., Tenn.
TRIBUTYL PHOSPHATE AS AN ALTERNATE SOLVENT FOR EXTRACTION OF "25" FROM SPENT MTR FUEL ASSEMBLIES. T. C. Runion. Sept. 26, 1950. Decl. Mar. 2, 1957. 12p. Contract W-7405-eng-26. $0.25(OTS).

The feasibility of a one-cycle solvent extraction process, using TBP, for the recovery and decontamination of U^{235} from spent MTR fuel assemblies was shown on a laboratory scale. (auth)

9661 ORNL-1138
Oak Ridge National Lab., Tenn.
THE HYDROLYSIS OF TRIBUTYL PHOSPHATE AND ITS EFFECT ON THE PUREX PROCESS. V. J. Reilly. Dec. 13, 1951. Decl. Mar. 2, 1957. 22p. Contract W-7405-eng-26. $0.30(OTS).

The rate of hydrolysis of TBP and the effect of the hydrolysis products in the Purex Process have been studied. Hydrolytic conditions may be encountered in the process which would lead to formation of dibutyl phosphoric acid, causing significant losses of Pu^{4+} in stripping. This situation may be easily alleviated by reducing and stripping the Pu^{3+}. (auth)

9662 ORNL-1314
Oak Ridge National Lab., Tenn.
A NEW ANIONIC SOLVENT EXTRACTION TECHNIQUE. F. L. Moore. July 17, 1952. Decl. Mar. 2, 1957. 9p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

The extraction of acids with long-chain amines is described. Preliminary results on the extraction of the amine salts of polonium, plutonium and uranium are given. (auth)

9663 ORNL-1338
Oak Ridge National Lab., Tenn.
ORGANO-PHOSPHORUS COMPOUNDS FOR SOLVENT EXTRACTION. C. E. Higgins, W. H. Baldwin, and J. M. Ruth. Aug. 14, 1952. Decl. Mar. 11, 1957. 17p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

The influence of the structure of phosphate esters on the distribution coefficient of U between solutions of these esters in CCl_4 and aqueous nitrate solutions is studied. The distribution coefficients of esters of secondary alcohols are compared with those of primary alcohols. The extraction efficiencies of alkyl and phenyl esters are compared, and the effect of negative groups on extraction is studied. The relationship of the number of C-P bonds on extraction is studied, and the distribution coefficients of tributyl phosphine oxide with uranyl nitrate, uranyl sulfate, uranyl chloride, Th nitrate, Pu(IV) nitrate, and fission products in nitrate solution are investigated. (W.L.H.)

9664 ORNL-1706
Oak Ridge National Lab., Tenn.
CHEMICAL SEPARATION OF ISOTOPES SECTION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1953. June 24, 1954. Decl. Mar. 1, 1957. 18p. Contract W-7405-eng-26. $3.30(ph OTS), $2.40(mf OTS).

The work on chemical separation of isotopes has been reactivated during this period. A proposed system for the separation of the isotopes of hydrogen and chlorine by means of tetrahydrofuran and hydrogen chloride has been investigated and has been shown to be chemically infeasible. Studies of the chlorine(gas) vs. chloride ion(aqueous) system have been initiated. Nitrous acid is presently being used as the refluxing agent in column studies of the isotopic separation factor. Results are reported on an exchange study in which Cl^{36} is used. No exchange of chlorine was found between chlorate and perchlorate ions in water solution after two years at room temperature. Hydrated lithium ions have been found to exchange all their molecules of solvation in a half-time of less than 0.8 min. Oxygen-18 was used as a stable isotopic tracer in this experiment. Methods of boron isotope separation have been examined. A liquid–liquid

system using boric acid and fluoboric acids distributed between an aqueous phase and an organic phase composed of a mixture of benzene and di(2- ethylhexyl) amine has been investigated, and results on distributions are presented. A gas-liquid type of system is proposed. Investigations are in progress on modifications of the ammonia(gas) vs. ammonium ion(aqueous) system for nitrogen isotope enrichment. An exchange column, stripping column, and associated apparatus have been built. Runs on this system show that calcium hydroxide slurry may be substituted for sodium hydroxide solution as the refluxing agent to convert ammonium ion to ammonia gas at the product (N^{15}) end of the system. This substitution should cut the cost of reflux chemicals considerably. A proposed closed-cycle system which would eliminate or greatly reduce the cost of reflux chemicals is under investigation. This system uses ammonium carbonate in place of ammonium nitrate in the aqueous solution. Initial difficulty has been encountered in plugging of the exchange column by a precipitate. Further runs are planned. (auth)

9665 ORNL-1979

Oak Ridge National Lab., Tenn.

DEVELOPMENT OF THE EXCER PROCESS. II. J. A. Marinsky. Apr. 17, 1956. Decl. Mar. 12, 1957. 35p. Contract W-7405-eng-26. $0.40(OTS).

Two revised Excer process flowsheets are presented for the production of UF_4 from $UO_2(NO_3)_2$ solution. In both, the U is sorbed on a cation-exchange resin and eluted with aqueous HF. In one, the resulting UO_2F_2 in dilute HF is mixed with dilute H_2SO_4 or HCl and electrolytically reduced in a cation-exchange-membrane cell, and the precipitated $UF_4 \cdot 0.75\ H_2O$ is dried and dehydrated to anhydrous UF_4. In the other, NH_4F is also added to the UO_2F_2 solution before the electrolytic reduction step, and the precipitate is $UF_4 \cdot 0.4\ NH_4F$. A flowsheet for preparing UF_4 from raw ore is also shown. The U is sorbed on an anion-exchange resin from a H_2SO_4 leach liquor and eluted with dilute HCl. Ammonium fluoride and HF are added to the UO_2Cl_2–HCl solution before the UO_2^{2+} is electrolytically reduced to precipitate UF_4 0.4 NH_4F. Decontamination factors as high as 45 and 9 for gross β and γ activities and of 10 to 5000 for light metal impurities were obtained in the electrolytic reduction step. (auth)

9666 ORNL-2227

Oak Ridge National Lab., Tenn.

FEED MATERIALS PROCESSING STATUS REPORT FOR NOVEMBER 1956. R. E. Blanco and J. C. Bresee. Feb. 13, 1957. Decl. Feb. 28, 1957. 9p. Contract W-7405-eng-26. $0.25(OTS).

The Fluorox Process UF_6 scrubbing column was operated unsuccessfully and is being modified. Fluidization of UF_4 with N_2 at 600°C was accomplished. In Excer Process development electrolytic reduction cells were studied. The 12-in. Higgins Contactor was tested on V ore with 99.8% recovery. (T.R.H.)

9667 TID-10022

Argonne National Lab., Lemont, Ill.

THE CHELATE PROCESS. II. H. W. Crandall, J. R. Thomas, and E. Zebroski. Aug. 17, 1947. Decl. Mar. 12, 1957. 19p. $0.25(OTS).

The Chelate Process for the extraction and decontamination of plutonium has been tested on a semi-works scale. Tests using C_6H_6 and $C_6H_5CH_3$ as the organic solvent gave an average of 98.7% Pu recovery for a three and a half cycle process. The average beta decontamination factors using C_6H_6 and $C_6H_5CH_3$ were $>10^7$ and 10^6 respectively. The average soft gamma decontamination factors using benzene and toluene were 10^6 and 10^4 respectively. The details of the semi-works experiments are given and the results of those experiments are critically evaluated. (T.R.H.)

SYNTHESES

9668 A-2191

Norton Co., Worchester, Mass.

THE DEVELOPMENT ON A COMMERCIAL SCALE OF A PROCESS FOR THE MANUFACTURE OF CRYSTALLINE BORON AND ANALYTICAL METHODS. Final Report. G. H. Fetterley and W. M. Hazel. Dec. 16, 1944. Decl. Feb. 14, 1957. 48p. For Columbia Univ. Contract W-7405-eng-50, Subcontract No. 7. (BXR-168). $7.80(ph OTS); $3.30(mf OTS).

A plant for the production of crystalline B of 95% purity was designed. A mixture of $BCl_3 + H_2$ is passed over a series of electrically heated graphite rods in a furnace. The B is removed from the rods, separated from adherent graphite and B carbide by sorting, and crushed and packaged. (C.H.)

9669 WAPD-CP-657

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

THE RADIATION SYNTHESIS OF AMMONIA IN A HIGH TEMPERATURE LOOP. Y Solomon, K. H. Vogel, and P. Cohen. Oct. 12 1954. Decl. Mar. 14, 1957. 36p. Contract AT-11-1-GEN-14. $6.30(ph OTS); $3.00(mf OTS).

A second NH_3 synthesis test has been run. Kinetics for the reaction consistent with the data of this test and past data are proposed. These kinetics involve a formation, first order in N_2, and a decomposition, first order in NH_3. Both reactions are directly proportional to the reactor power. The effect of using non-deaerated make-up water is evaluated. It is concluded that, from the point of view of O_2 removal and NH_3 synthesis, non-deaerated make-up appears feasible. An evaluation of the feasibility of NH_3 as a pH control method is made, with the conclusion that it is of sufficient interest to warrant testing at a higher NH_3 concentration. (auth)

URANIUM AND URANIUM COMPOUNDS

9670 A-1253

Columbia Univ., New York. Div. of War Research.

SPECTROSCOPIC DATA OF URANYL AND URANOUS SALTS. R. H. Crist. Feb. 15, 1944. Decl. Mar. 21, 1957. 150p. Project SAM. Contract W-7405-Eng-50. (2R-1241). $22.80(ph OTS); $7.20(mf OTS).

9671 A-3719

Harshaw Chemical Co., Cleveland.

EQUIPMENT AND OPERATING CONDITIONS. PROPOSED PROCESS FOR THE RECOVERY OF C-616 [UF_6]. H. W. Cromer and C. F. Swinehart. Apr. 10, 1945. Decl. Feb. 12, 1957. 10p. For Kellex Corp. Contract W-4705-eng-23, Subcontract No. 38. $1.80(ph OTS); $1.80(mf OTS).

9672 ACCO-14

American Cyanamid Co. Atomic Energy Div., Watertown, Mass.

CALCINING-AMENABILITY TESTING OF RAND PRECIPITATES. F. W. Bloecher, Jr. Nov. 26, 1951. Decl. Mar. 16, 1957. 15p. Contract AT(49-1)-533. $3.30(ph OTS); $2.40(mf OTS).

High-grade Rand U precipitates were calcined at 400 to 500°C. The precipitate which had been dried or calcined at any temperature up to 500°C, was found to be amenable to treatment at a refinery by the HNO_3 digestion, TBP process. Calcining was found to be desirable in order to effect savings in shipping costs. The percentage of U_3O_8 left in the final residue was used as the index for judging amenability of the calcines to the refinery treatment process. (L.D.)

9673 CF-56-6-138
Oak Ridge National Lab., Tenn.
RECOVERY AND PURIFICATION OF URANIUM BY CONTINUOUS ION EXCHANGE AND ELECTROLYSIS. I. R. Higgins. June 26, 1956. Decl. Feb. 15, 1957. 19p. Contract [W-7405-eng-26]. $0.30(OTS).

Uranium recovery and purification by continuous ion exchange and electrolysis, either independently or jointly in a single process, are being studied. The function of the ion exchange step is to concentrate, partially purify, and prepare uranium in a suitable chemical form for the subsequent purification step. The function of electrolysis is to provide more purification and reduce UO_2F_2 to $UF_4 \cdot {}^3/_4 H_2O$. A number of chemical process steps are used for uranium processing, whether from ores to concentrates to pure metal or hexafluoride, or from spent reactor fuels back to purified uranium. The ion exchange and electrolysis steps are proposed as shorter and less expensive routes in this uranium cycle. Even these steps may be complicated by attempts to adapt them to the chemistry of existing processes. (auth)

9674 HCC-1
Harshaw Chemical Co., Cleveland.
CONVERSION OF BROWN OXIDE TO GREEN SALT WITH HYDROGEN FLUORIDE AND HYDROFLUORIDE ACID. Progress Report for Period March 1, 1948 to June 1, 1948. J. B. Klingel. July 20, 1948. Decl. Feb. 19, 1957. 57p. Contract W-7405-Eng-276. $0.50(OTS).

This report describes in detail a number of experiments on bench scale and semi-plant scale wherein the effect of various variables upon the rate of reaction between HF and UO_4 have been determined. The purpose of this work was to determine the effect of these variables on the speed of reaction in order that an improved reactor system might be developed for the production of UF_4 from UO_2 which would require less labor, less plant floor space and decrease the health hazards due to exposure and to dusting. The experimental results indicated the optimum conditions for the conversion of UO_2 to UF_4 with HF to be as follows: (1) temperature 450 to 550°C; (2) HF concentration must be in excess of 90% by weight at the end of the reaction in order to produce a product which will analyze better than 95% UF_4; (3) UO_2 bed thickness must be held to a minimum; (4) it is practical to use large scale rotary reactors. (A.M.)

9675 HW-26531
Hanford Works, Richland, Wash.
PRODUCTION OF UO_3 BY CALCINATION OF URANYL PEROXIDE. R. L. Moore and R. A. Watts, Jr. Dec. 10, 1952. Decl. Feb. 28, 1957. 3p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

9676 KLX-1208
Kellex Corp., New York.
PROGRESS REPORT [FOR] OCTOBER 1950. Job 23. Nov. 10, 1950. Decl. Feb. 27, 1957. 47p. Contract AT(30-1)-848. $7.80(ph OTS); $3.30(mf OTS).

Certain revisions have been made in the Job 23 research program. The search for a mechanically suitable MgX leaching system has been continued. Column runs were completed with both C-3 discard liquor, using aminized cotton cloth, and neutralized C-3 discard liquor, using IRA-400. The investigation of the recovery of uranium from Tonawanda solid waste has been resumed. A Scheibel extraction column and a glass-bead-packed scrubbing and strip column have been erected and tested for use in a solvent comparison study. Five-stage batchwise extraction of the supernate resulting from the treatment of MCW Raffinate II with an equal volume of water-saturated diethyl ether has been completed. (B.J.H.)

9677 MCW-6
Mallinckrodt Chemical Works, St. Louis.
THE PREPARATION OF URANIUM TRIOXIDE FROM URANYL NITRATE SOLUTION. Harold Yeager. Apr. 1, 1946. Decl. Feb. 26, 1957. 11p. Contract W-7405-Eng-1. $0.25(OTS).

Laboratory experiments on conversion of pure $UO_2(NO_3)_2$ to UO_3 by (1) thermal decomposition and (2) precipitation from solution with NH_3 or H_2O_2 are reported. Method (1) was chosen for plant experiments, pertinent data on which are presented. It is concluded that successful conversion can be accomplished in gas-fired type-347 stainless steel kettles with proper agitation to break up the hard cake which forms. By controlled subsequent heating, UO_3 can be obtained which contains only a trace of water, nitrate, and U_3O_8. Continued heating of the powder may be sufficiently controlled by the stack temperature of the burning gases. (G.Y.)

9678 MCW-144
Mallinckrodt Chemical Works, St. Louis.
FLUORINATION OF URANIUM TRIOXIDE-RATE AND "APPARENT EQUILIBRIUM" STUDIES. Carl W. Kuhlman, Jr. Oct. 19, 1948. Decl. Feb. 26, 1957. 8p. $1.80(ph OTS); $1.80(mf OTS).

The rates of the reaction $UO_3 + 2HF \rightarrow UO_2F_2 + H_2O$ were studied in the temperature range from 90 to 500°C employing both anhydrous HF and mixtures of HF and H_2O. This reaction was found to proceed quite rapidly even at low temperatures and with dilute $HF-H_2O$ mixtures being employed These rates may be described by the following equation if the vapor phase consists of anhydrous HF. An attempt was made to measure the equilibrium concentrations for the reactions, but owing to the difficulty in reversing the reaction, only approximate values could be obtained for Kp. (G.Y.)

9679 MCW-179
Mallinckrodt Chemical Works, St. Louis.
UF_4 PRODUCTION BY CONTINUOUS METHODS. Progress Report of Pilot Plant Work for Period January 1, 1949 to March 4, 1949. E. K. Teter. Apr. 11, 1949. Decl. Feb. 27, 1957. 12p. $3.30(ph OTS); $2.40(mf OTS).

Uranium dioxide was hydrofluorinated in two types of 8-in. I.D. gas-heated tubes fitted with 4-in.-pitch conveyers. With respect to lumpiness of product, yield, and tube deterioration, an Inconel tube was better than a Mn-lined steel tube, and an Inconel ribbon screw conveyor was better than a calorized steel paddle converyer. Gas heating did not prove too satisfactory because of the large temperature difference between top and bottom of the tube. Green salt (UF_4) that met specifications could be produced in this reactor. (G.Y.)

9680 MCW-191
Mallinckrodt Chemical Works, St. Louis.
PRODUCTION OF UO_3 FROM SOLID $UO_2(NO_3)_2 \cdot XH_2O$. A. M. Kirby. June 15, 1949. Decl. Feb. 26, 1957. 8p. Contract [W-14-108-Eng-8]. $0.25(OTS).

Two methods are outlined for producing UO_3. The first consisted of heating molten hexafluoride to about 225°C so that a solid product was obtained upon cooling to about 150°C and then decomposing at this temperature in vacuo. The second method consisted of dehydrating $UO_2(NO_3)_2 \cdot 3H_2O$ at 110 to 120°C with a stream of CO_2 containing HNO_3 vapors and then decomposing at about 160°C in vacuo. The behavior of the material in both cases was much the same. The desired product was finally obtained by prolonged heating at 160°C in vacuo using the second method. A diagram of the apparatus for the decomposition of solid $UO_2(NO_3)_2 \cdot XH_2O$ is shown. (D.E.B.)

9681 MCW-209
Mallinckrodt Chemical Works, St. Louis.
UF_4 PRODUCTION BY CONTINUOUS METHODS. Progress Report of Pilot Plant Work [for] June 19, 1949 to July 13, 1949. E. K. Teter. July 29, 1949. Decl. Feb. 26, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

Progress of the work on continuous UF_4 production using various type screw conveyers at various operating conditions is presented. The so-called compound screw (double flight flat spiral with deflectors) was found to be the best conveyer yet tried. It yielded better fluorination per pass with a finer, lump-free, product. (J.E.D.)

9682 MCW-213
Mallinckrodt Chemical Works, St. Louis.
"INHIBITION" OF THE OXIDATION OF LOW-TEMPERATURE URANIUM DIOXIDE. Carl W. Kuhlman, Jr. Aug. 23, 1949. Decl. Feb. 26, 1957. 4p. $0.20(OTS).

Uranium dioxide prepared by the reduction of UO_3 at low temperature was found to be very reactive toward HF. In the preparation of UO_2 for use in the UF_4 pilot plant, difficulty was experienced in obtaining a product of high UO_2 assay due to the extremely rapid rate of oxidation by air of the freshly reduced dioxide. An account of attempts to find a method of "inhibiting" this oxidation reaction without disturbing the rate of the subsequent fluorination is given. HF treatment was found to inhibit oxidation quite well; NH_3, N_2, CCl_4, ethyl alcohol, and toluene were less successful. (auth)

9683 MCW-214
Mallinckrodt Chemical Works, St. Louis.
A NOTE ON THE FLUORINATION OF U_3O_8 WITH ANHYDROUS HYDROGEN FLUORIDE. Carl W. Kuhlman, Jr. Aug. 26, 1949. Decl. Feb. 26, 1957. 4p. $0.20(OTS).

Experiments were performed to determine the behavior of the intermediate oxide U_3O_8 during treatment with HF. U_3O_8 was found to fluorinate much more rapidly than UO_2 at 540°C to give 2 moles of UO_2F_2 and 1 mole of UF_4. Thus in plant equipment it may be concluded that $^2/_3$ of any U_3O_8 introduced into a fluorination apparatus will probably "report" in the water-soluble fraction of the product. (D.E.B.)

9684 MCW-215
Mallinckrodt Chemical Works, St. Louis.
REDUCTION OF URANIUM TRIOXIDE WITH HYDROGEN–NITROGEN MIXTURES. Carl W. Kuhlman, Jr. Sept. 2, 1949. Decl. Feb. 26, 1957. 6p. $0.25(OTS).

Kinetic measurements of UO_3 reduction were made by passing H over dried samples at 1 l/min. The partial pressure of H was controlled by introducing purified N. The ratio of the flow rates of H and N determined the partial pressure of H. Total pressure was maintained at 1 atm. Reduction rates are tabulated for 0.2, 0.5, and 0.75 atm H partial pressure at 450°C, and for 0.5, 0.2, and 0.1 atm at 500°C. Utilization of a H–N mixture with H partial pressure of 0.75 atm corresponds to utilization of cracked NH_3. At temperatures below 500°C a definite retardation is produced by the introduction of N. (D.E.B.)

9685 MCW-217
Mallinckrodt Chemical Works, St. Louis.
REDUCTION OF URANIUM TRIOXIDE TO URANIUM DIOXIDE WITH ANHYDROUS AMMONIA. Carl W. Kuhlman, Jr. Sept. 8, 1949. Decl. Feb. 26, 1957. 7p. $0.25(OTS).

Rates of UO_3 production to UO_2 by NH_3 were measured at 450, 499, 534, and 600°C using solid layers of dried UO_3 and excesses of NH_3. Velocity constatns calculated were lower by about a factor of 10 than those for H_2 reduction. Apparent activation energies for the 2 reactions differed by 10%. The difference in reaction rates may be explained by H partial pressure assuming H and not NH_3 to be the effective reducing agent. (D.E.B.)

9686 MCW-220
Mallinckrodt Chemical Works, St. Louis.
UF_4 PRODUCTION BY CONTINUOUS METHODS. Progress Report of Pilot Plant Work [for] Period July 14, 1949 to August 16, 1949. E. K. Teter. Aug. 31, 1949. Decl. Feb. 26, 1957. 15p. Contract [W-14-108-eng-8]. $3.30 (ph OTS); $2.40(mf OTS).

Progress of the work on the continuous UF_4 production using varying conditions for a compound conveyor and an angle conveyor, and a study was made of the effect of using an H_2HF gas mixture for the reacting gas. It was concluded that the compound conveyer gave better fluorination per pass than the angle conveyer, and that because of the greater stirring action and longer retention possible with the compound conveyer it was felt that with a larger and more complete pilot plant, complete fluorination can be accomplished in "one pass." (J.E.D.)

9687 MCW-225
Mallinckrodt Chemical Works, St. Louis.
RATE OF DECOMPOSITION OF URANIUM TRIOXIDE ON HEATING. Carl W. Kuhlman. Oct. 7, 1949. Decl. Feb. 26, 1957. 10p. $0.25(OTS).

The decomposition rate was determined in a $^1/_2$-in. tube furnace equipped with a quartz tube and regulated by a 2 KVA powerstat used to heat the sample. Temperatures inside the tube were measured with a chromel alumel thermocouple. Water pumped N served as a sweeping gas. The measured loss in weight was corrected for water and oxides of N lost in the preliminary heating by the subtraction $\Delta W = \Delta W_m - 0.1717$. An outline of the procedure is given, and results are tabulated. (J.E.D.)

9688 MCW-227
Mallinckrodt Chemical Works, St. Louis.
DISTRIBUTION OF BORON BETWEEN THE ETHER PHASE AND WATER PHASE IN URANYL NITRATE SOLUTIONS. C. W. Kuhlman and Walter Weber. Oct. 17, 1949. Decl. Feb. 27, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

The object of these experiments was to determine the partition coefficient of B in water and ether solutions of U concentrations which are used in the neutral wash system and of B concentrations which may be encoutered in the plant. A complete pilot-plant, ether-extraction run was made using an acid feed spiked with boric acid. A portion of the neutral ether extract was washed successively six times, and the aqueous and ether phases were both analyzed for U and B content. The partition coefficient of B at a U concentration of about 320 g/l in the ether phase and in a B concentration range of 0.12 to 8.4 ppm B/U in the ether phase was found to average 0.056 for the ratio of the concentration of B in the ether phase to the concentration of B in the aqueous phase. (A.M.)

9689 MCW-246
Mallinckrodt Chemical Works, St. Louis.
UF_4 PRODUCTION BY CONTINUOUS METHOD. Progress Report of Pilot Plant Work [for] Period November 20, 1949 to January 5, 1950. E. K. Teter. Jan. 27, 1950. Decl. Feb. 27, 1957. 18p. $3.30(ph OTS); $2.40(mf OTS).

Satisfactory conversion of UO_2 to UF_4 was maintained in an electrically heated, double-pass, Inconel reactor. At optimum temperatures and HF rate UO_2 could be fed at a rate of 100 lb/hr. Ni contamination caused by abrasion was reduced by reducing the speed of the conveyors. There was also evidence that the amount of HF charged in excess of the amount needed to complete the reaction might effect Ni contamination. Each reactor was independent of any other reactor so that stresses caused by expansion of one

reactor were not transmitted to another. Conveyors were driven so that material was pushed rather than pulled through each reactor. In these runs, the conveyor shaft and gear reducer output shaft were connected by a double universal to allow for reactor expansion and side to side movement. (D.E.B.)

9690 MCW-249

Mallinckrodt Chemical Works, St. Louis.

CORROSION TESTS LEADING TO THE RECOMMENDATION OF ILLIUM R FOR GREEN SALT (UF_4) REACTORS. C. F. Ritchie and E. K. Teter. Mar. 14, 1950. Decl. Feb. 26, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

The alloys copel, nichrome-V, monel 326, stainless steel-588, and illium-R have been tested for embrittlement and for scale formation and adhesion after having served as reactor boats in which UF_4 was prepared. Details regarding these five are listed for record. Illium-R proved a surprising and outstanding alloy. It took on a light scale immediately, but this scale neither "grew" appreciably nor flaked off. Resistance to embrittlement seems satisfactory. A recommendation based on these findings is made that illium-R sheet and bar stock be used for tube and screw construction in the continuous UF_4 plant. (G.Y.)

9691 NYO-1135

Vitro Mfg. Co., Pittsburgh.

PROBLEMS OF LEACHING AND DIGESTION OF URANIFEROUS SLAGS. Progress Report No. 1 [Covering] Period April–May 1951. George Toelken. July 1, 1951. Decl. Feb. 28, 1957. 8p. Contract AT(30-1)-1134. $1.80(ph OTS); $1.80(mf OTS).

Time, temperature, and screening requirements for satisfactory hot and cold digestion of U-bearing slag were determined in laboratory and plant tests. In some cases H_2 was evolved rapidly enough to present the danger of spontaneous ignition. Laboratory tests indicated the F cannot be removed satisfactorily by clarification of the digest liquors, complete washing of the crude $(NH_4)_2UO_4$ precipitate, or reprecipitation of the uranate after digestion in dilute H_2SO_4. A Na_2CO_3 separation followed by precipitation of Na_2UO_4 with NaOH gave a satisfactory product. (D.E.B.)

9692 NYO-1150

Vitro Mfg. Co., Pittsburgh.

PROBLEMS OF REFINING URANIFEROUS RESIDUES; TECHNOLOGY — FEED MATERIALS. Progress Report No. 16 [for] August 1952. H. Fleck and J[ohn E.] Summers. Sept. 17, 1952. Decl. Feb. 28, 1957. 29p. Contract AT(30-1)-1241. $4.80(ph OTS); $2.70(mf OTS).

An investigation of the applicability and refinement of various procedures for the treatment and digestion of carbonate and other uraniferous residues for recovery of U is presented. (F.S.)

9693 NYO-1304

Mallinckrodt Chemical Works, St. Louis.

PILOT PLANT WORK ON UF_4 PRODUCTION BY CONTINUOUS METHODS. Progress Report [for] Period March 6, 1950 to April 21, 1950. E. K. Teter. May 15, 1950. Decl. Feb. 28, 1957. 9p. $0.25(OTS).

UO_2 was fluorinated in a double pass Inconel reactor, the material being conveyed by Inconel conveyors. It was found that the pitch of the deflectors on the conveyors determines whether or not lumps would be formed. A Monel liner was installed in the upper reactor but during a run the conveyor caught on the liner seam and twisted. The Adams poro-carbon filter has filtered the reactor exhaust gases very well. The filter can be easily cleaned by backblowing the tubes with a small amount of air at low pressure. (auth)

9694 NYO-1329

Mallinckrodt Chemical Co., St. Louis.

PRODUCTION OF URANIUM METAL FROM URANIUM TETRAFLUORIDE MADE FROM URANIUM TRIOXIDES PRODUCED AT HARSHAW CHEMICAL CO. W. H. Keller. Dec. 27, 1951. Decl. Mar. 2, 1957. 8p. $0.25(OTS).

Uranium(VI) oxide produced batchwise in the thermal decomposition pots was reduced to UO_2, hydrofluorinated to UF_4, reduced with Mg to metal, and cast. This product produced satisfactory intermediates, high metal reduction yields (average 98.2%), and good quality metal in casting. A second type of UO_3 produced on a drum dryer was converted to metal by a similar procedure. This UO_3 was light in weight and of a porous flaky structure. The UF_4 was of low density and the yields in the metal reduction step were considerably lower. (W.L.H.)

9695 NYO-5000

Harshaw Chemical Co., Cleveland.

PRODUCTION OF (URANIUM) TETRAFLUORIDE. K. E. Long. Apr. 8, 1943. Decl. Feb. 26, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

The production of UF_4 by the dry reaction of UO_2 with HF at 550°C is described and discussed. (T.R.H.)

9696 NYO-5027

Mallinckrodt Chemical Works, St. Louis.

REPORT ON AMMONIUM BIFLUORIDE AS A [HYDRO] FLUORINATING AGENT FOR BROWN OXIDE [UO_2]. D. Copenhafer. Jan. 30, 1943. Decl. Feb. 26, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

The practicability of converting brown oxide (UO_2) to green salt (UF_4) by the action of $NH_4F \cdot HF$ was studied. At 150 to 200°C and an excess of 50 mole % $NH_4F \cdot HF$, conversion was not more than 40% unless considerable H_2O was present; the purity of the green salt thus obtained was poor. Conversion at 500°C was rapid and, with a 50 mole % excess of bifluoride, about 94%. This yield increased to 99.1% on the addition of anhydrous HF. Consistent low moisture and NH_3 contents were noted in the 500°C experiments. Hot constant-boiling HF(43.2%) did not convert UO_2 to UF_4 even after considerable time. (J.E.D.)

9697 NYO-5106

Mallinckrodt Chemical Works, St. Louis.

A STUDY OF SOME INTERFERING ELEMENTS IN THE VOLUMETRIC ASSAY OF X_3O_8 [U_3O_8]. J. R. McCoy. April 17, 1945. Decl. Mar. 8, 1957. 8p. Contract W-7405-eng-1. $1.80(ph OTS); $1.80(mf OTS).

The mechanisms of the interference of Ti, Mo, V, and Fe in the volumetric determination of U_3O_8 with ceric sulfate were investigated. The original theories and the experimental data are presented, but no conclusions or theories based on the data are included. The interfering elements were studied by means of the following compounds: $K_2TiF_6 \cdot H_2O$, MoO_3, NH_4VO_3, and $FeNH_4(SO_4)_2 \cdot 12H_2O$. (A.T.M.)

9698 NYO-5127

Mallinckrodt Chemical Works, St. Louis.

A STUDY OF CRACKED AMMONIA CONSUMPTION EFFICIENCY IN THE REDUCTION OF XO_3 [UO_3]. James C. Graham. June 15, 1945. Decl. Mar. 7, 1957. 13p. $0.30 (OTS).

Due to the fact that the consumption of cracked NH_3 for the reduction of UO_3 to UO_2 was increasing, experiments were undertaken to determine the cause and to determine how the process should be run to realize the greatest efficiency from the cracked NH_3. Plots were made to show the relationship of the amount of NH_3 used to the average assay of the material produced on the milled cake runs and also to determine the relationship of the peak pressure used during the cycle to the assay of the finished product.

Plots of the data taken did not give conclusive results. (J.E.D.)

9699 NYO-5228
Mallinckrodt Chemical Works, St. Louis.
THE DETERIORATION OF UO_2 IN STORAGE. G. L. Martin. May 1948. Decl. Mar. 7, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

Long-term storage of UO_2 in non-airtight containers will result in losses from 2 to 4% due to oxidation as a result of temperature effects. Losses could be reduced by producing UO_2 with smaller specific surface. (W.L.H.)

9700 TID-5343
Vitro Mfg. Co., Pittsburgh.
FLOW SHEETS FOR C-LINER PROJECT. Supplement. A. J. Strod. Aug. 6, 1951. Decl. Mar. 7, 1957. 5p. (M-5483). $1.80(ph OTS); $1.80(mf OTS).

Three practical procedures for the extraction of U from slag were developed, and flow diagrams are given. These processes are designated as dry raw, dry roasted, and wet raw. (B.J.H.)

ENGINEERING

9701 CF-56-8-102
Oak Ridge National Lab., Tenn.
DESIGN STUDY—HRT PRESSURIZER LEVEL FLOAT. C. L. Segaser. Aug. 8, 1956. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Float weight vs. length was plotted for 6-in.^{3} displacement floats constructed of stainless steel and titanium. (D.E.B.)

9702 CF-56-9-34
Oak Ridge National Lab., Tenn.
SUGGESTED DESIGN PRACTICES FOR STEEL REINFORCING AND FOR EXPANSION AND CONTRACTION JOINTS IN CONCRETE SHIELDING FOR NUCLEAR REACTORS. H. L. Kinnier. Sept. 10, 1956. 16p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

HEAT TRANSFER AND FLUID FLOW

9703 CF-56-7-59
Oak Ridge National Lab., Tenn.
A METHOD OF CORRELATING EXPERIMENTAL FLUID FRICTION DATA FOR SQUARE ARRAY TUBULAR HEAT EXCHANGERS. J. L. Wantland. July 18, 1956. 10p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

An analytical method of correlating fluid friction data for flow parallel to square array cylindrical tubes in rectangular cross-sectional flow channels has been devised. Using this technique, the experimental values for friction factors and Reynolds moduli are multiplied by correlation terms which yield new friction factors and Reynolds moduli that pertain to the core region of the tube bundle excluding the wall effects. Model configurations with different numbers of tubes and different tube bundle to shell wall spacing can thus be correlated by converting the data to those which would apply to an infinite array tube bundle. (auth)

9704 CF-56-9-131
Oak Ridge National Lab., Tenn.
PROPOSED SYSTEM FOR ATMOSPHERIC PRESSURE SLURRY HEAT TRANSFER STUDIES. D. G. Thomas. Sept. 13, 1956. 13p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

9705 CF-56-10-23
Oak Ridge National Lab., Tenn.
FLUID MOTION IN "U"-TUBES OF VARIABLE CROSS-SECTION (HRT LEVEL INDICATORS). R. F. Hughes. Oct. 2, 1956. 14p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

Equations are developed which describe the oscillatory motion of a fluid in a "U" tube resulting from an initial unbalance. (D.E.B.)

9706 KT-97
Massachusetts Inst. of Tech., Oak Ridge, Tenn. Engineering Practice School.
THE DESIGN, CONSTRUCTION, AND TESTING OF A SYSTEM FOR THE STUDY OF BUBBLE FORMATION AT HIGH POWER DENSITIES. R. Bakal and D. L. James. Oct. 23, 1950. Decl. Feb. 28, 1957. 19p. For Carbide and Carbon Chemicals Div. [K-25 Plant. Contract W-7405-eng-26, Subcontract 70]. $4.80(ph OTS); $2.70(mf OTS).

MATERIALS TESTING

9707 CF-56-10-27
Oak Ridge National Lab., Tenn.
NON-DESTRUCTIVE TESTS ON 2½" TITANIUM PIPE AND HOUSING FOR CENTRIFUGAL PUMP. Joseph P. Hammond and T. M. Kegley, Jr. Oct. 4, 1956. 4p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

PUMPS

9708 A-2369
[Standard Oil Co. of Indiana, Chicago.]
EXPERIENCE IN THE USE OF DIAPHRAGM PUMPS. T. G. Stack and J. J. Lukes. July 5, 1946. Decl. Feb. 12, 1957. 30p. Contract W-7418-eng-41. $6.30(ph OTS); $3.00(mf OTS).

Diaphragm pumps of a type manufactured by Wilson Chemical Feeders, Inc. were operated under vacuum in an automatic flow control system and were found to perform satisfactorily in a continuous unit for more than a year. In order to make operation under vacuum possible, it was necessary to use a spring to move the diaphragm on the suction stroke. Other modifications which were made included the use of larger diaphragm heads with larger bolts, the development of full-faced diaphragm gaskets, and the substitution of disk type suction check valves for those normally used. The pump was adapted to automatic flow control by the use of a pressure regulating valve which permitted by-passing of excess flow. In order to secure high volumetric efficiency, it was found that care must be taken to eliminate air leakage and the accumulation of air in the hydraulic system of the pumps. Considerable difficulty in breakage of diaphragms was encountered at an early stage of the project, but the difficulty was eliminated by redesigning the enclosing curved surfaces so as to help prevent the diaphragm from being stressed beyond its fatigue limit. Although attempts were made to develop a theoretical method for designing the curvature of these surfaces, it was found best to use data obtained experimentally. (auth)

9709 CF-56-8-120
Oak Ridge National Lab., Tenn.
HRT-CP ENGINEERING TEST PROCEDURE P-1; WESTINGHOUSE CIRCULATING PUMP. P. E. Novak. Aug. 15, 1956. 19p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

Performance tests were conducted on canned-rotor

pumps for the HRT fuel processing system. A description of the tests and their results are given. (D.E.B.)

9710 CF-56-8-149
Oak Ridge National Lab., Tenn.
EXAMINATION OF TWO 347 SS DIAPHRAGMS FROM PULSE-FEEDER PUMPS. Bertram Fleischer. Aug. 23, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Two 347 SS pump diaphragms (20 mils thick) from pulse-feeder pumps were submitted to metallography for examination. One of the diaphragms had failed after 2800 hrs in the HRT mock-up. This diaphragm was exposed to 0.04 molal UO_2SO_4 on one side and water on the other. The temperature was approximately 150°F. The other diaphragm was exposed (700 hrs.) to water on both sides in a pulse-feeder test loop. No failure had occurred. (auth).

MINERALOGY, METALLURGY, AND CERAMICS

9711 BMI-1121(Del.)
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS RELATING TO CIVILIAN APPLICATIONS DURING JULY 1956. Russell W. Dayton and Clyde R. Tipton, Jr. Aug. 1, 1956. Decl. with deletions Feb. 12, 1957. 41p. Contract W-7405-eng-92. $7.80(ph OTS); $3.30(mf OTS).

Research results are reported in temperature distribution in breeding blanket rods, calcination of BeO powder, tensile properties of U–Mo alloys at high temperatures, mechanism of U corrosion by water, and corrosion of U–Zr alloy by water at high temperatures. (T.R.H.)

9712 FMPC-296
National Lead Co. of Ohio, Cincinnati.
A FURNACE BUILT TO CONVERT URANIUM METAL TURNINGS TO U_3O_8. A. E. Guay and H. M. Eikenberry. July 15, 1953. Decl. Feb. 20, 1957. 20p. Contract AT(30-1)-1156. $0.25(OTS).

The design, construction, and initial operating period of a furnace built to convert U turnings to U_3O_8 are described. The furnace has a design capacity of 300 lbs of U turnings per hr. (D.E.B.)

CERAMICS AND REFRACTORIES

9713 BMI-1074(Del.)
Battelle Memorial Inst., Columbus, Ohio.
REFRACTORIES FOR CONTINUOUS REDUCTION OF URANIUM. Allison K. Smalley, Collin Hyde, Adrian G. Allison, and Winston H. Duckworth. Feb. 27, 1956. Decl. with deletions Feb. 13, 1957. 8p. Contract W-7405-eng-92. $1.80(ph OTS); $1.80(mf OTS).

A magnesia refractory was resistant to both molten NaF and molten U metal at 2200°F in short-time tests. It was attacked moderately by molten AlF_3 at 2100°F and severely by molten MgF_2 and CaF_2 at 2600 and 2700°F, respectively. Uranium metal at 3000°F also attacked it severely. Magnesia refractories appear to warrant further consideration as a lining material in reaction vessels for the continuous reduction of UF_4 by Na, provided that the refractory temperatures can be maintained below 3000°F. Refractories made of alumina, stabilized zirconia, mullite, chrome-magnesite, or alumina-chromite were considered unsuitable for this application, owing to the ease with which they were attacked by NaF, MgF_2, CaF_2, and AlF_3. (auth)

9714 KAPL-M-RSM-1
Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT OF TRIP TO WESTINGHOUSE, BETTIS FIELD PLANT, FEBRUARY 15, 1956. R. S. Miller and J. N. Frank. Feb. 23, 1956. Decl. Mar. 5, 1957. 9p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

A detailed outline of PWR oxide fuel element fabrication is given. (D.E.B.)

CORROSION

9715 ANL-4862
Argonne National Lab., Lemont, Ill.
AQUEOUS CORROSION OF URANIUM AND ALLOYS: SURVEY OF PROJECT LITERATURE. J. W. McWhirter and J. E. Draley. May 14, 1952. Decl. Feb. 7, 1957. 49p. Contract W-31-109-eng-38. $0.40(OTS).

The corrosion rate of U in H_2-saturated and air-saturated water is reported. The effects of dissolved gas, H_2O_2, dissolved salts, velocity, surface preparation, and heat-treatment on aqueous corrosion of U are discussed. Aqueous corrosion of Nb–U, Mo–U, Si–U, U–Zr, and Al–U alloys was studied, and the effect of alloying constituents on corrosion rate is reported. (auth)

9716 ANL-5147(Rev.)
Argonne National Lab., Lemont, Ill.
CORROSION AND STABILITY TESTS ON CHEMICAL POISONS IN HIGH-TEMPERATURE WATER. C. R. Breden and Alma Abers. Sept. 1, 1953. Decl. Mar. 20, 1957. 48p. Contract W-31-109-Eng-38. $0.45(OTS).

Corrosion-stability tests have been made in static autoclaves at 500 and 600°F on solutions of compounds having high neutron cross sections to evaluate their usefulness for shutdown purposes. The only compound tested which appeared to be completely stable in 600°F water was H_3BO_4. Limited corrosion data did not show it to cause excessive corrosion of Zr or stainless steel. (auth)

9717 ANL-5196
Argonne National Lab., Lemont, Ill.
PRELIMINARY REPORT ON CORROSION OF LOW-URANIUM, ZIRCONIUM-BASE ALLOYS. A. E. Dwight and A. H. Roebuck. Oct. 1953. Decl. Feb. 12, 1957. 31p. Contract W-31-109-Eng-38. $0.30(OTS).

Tests were made to determine the effects of heat treatment and composition on the corrosion resistance of low-uranium, zirconium-base alloys to water at 600°F. A total of 57 compositions were tested. The zirconium alloys contained up to 9% natural uranium plus small amounts of tin, antimony, lead, bismuth, yttrium, beryllium, germanium, niobium, nickel, and aluminum in various combinations. Data are presented in both tabular and graphical form. The effect of heat treatment on corrosion resistance of zirconium–uranium–tin alloys is partially masked by impurities in alloys made from sponge or bomb-reduced zirconium. In ternary alloys made from crystal bar zirconium, the effect of heat treatment on corrosion resistance is definite, and varies with the composition of the alloy. The range of ternary compositions from approximately 2 to 6 weight-per cent uranium and from 2 to 4 per cent tin, are the most attractive fuel element core alloys. (auth)

9718 ANL-RCV-SL-1090
Argonne National Lab., Lemont, Ill.
DISSOLUTION OF THORIUM OXIDE–URANIUM OXIDE FUEL ELEMENTS. R. P. Larsen. Aug. 9, 1956. Decl. Mar. 9, 1957. 2p. Contract [W-31-109-eng-38]. $1.80 (ph OTS); $1.80(mf OTS).

9719 CF-56-7-56
Oak Ridge National Lab., Tenn.
METALLOGRAPHIC EXAM. OF TYPE 347 SS AND TITANIUM 75A CORROSION SPECS. EXPOSED TO VAPOR ABOVE OXYGENATED 0.14 m UO_2SO_4 CONTAINING CHLORINE OR IODINE. METALLORGRAPHIC REPORT NO. 21 (Y-12). T. M. Kegley, Jr. July 16, 1956. 53p. Contract [W-7405-eng-26]. $9.30(ph OTS); $3.60(mf OTS).

Type 347 stainless steel and titanium 75A specimens were received for metallographic study. These specimens were selected from a group of specimens which had been subjected to vapor phase exposure above oxygenated 0.14m UO_2SO_4 solutions containing varying amounts of chlorine or iodine. The test conditions are described briefly in the accompanying table. The specimens were first examined with a low power microscope and appropriate photomacrographs made of the specimen surfaces. Sections of each of the specimens were then taken for metallographic examination. The metallographic results are presented in photographs and a table. (auth)

9720 CF-56-9-59
Oak Ridge National Lab., Tenn.
SUMMARY OF TOROID RUN #92: TYPE 347 STAINLESS STEEL, TITANIUM 75A AND ZIRCALOY-2 IN THORIUM OXIDE SLURRIES AT 250° AND 26 fps. EFFECT OF $Na_4P_2O_7$ ADDITIONS AND HYDROGEN ATMOSPHERES. G. E. Moore and E. L. Compere. Sept. 8, 1956. 15p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

9721 CF-56-10-37
Oak Ridge National Lab., Tenn.
EXAMINATION OF RECOMBINER LOOP SPECIMENS OF Zr AND Ti ALLOYS. M. L. Picklesimer. Oct. 2, 1956. 13p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40 (mf OTS).

Specimens of Ti, Ti alloy, Zr, and Zr alloy were corrosion tested in an $H_2 - O_2$ recombiner loop. The results of hydridation and grain growth examinations were inconsistent. It is recommended that the tests be repeated under more controlled conditions. (D.E.B.)

9722 CF-56-11-138
Oak Ridge National Lab., Tenn.
EVALUATION OF VARIOUS OXIDES IN 100-A LOOPS (SUMMARY OF RUNS S-58, 59, 64, 65, 66, 84, 85, 90). A. S. Kitzes, R. B Gallaher, and C. A. McLaughlin. Nov. 30, 1956. 21p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

It was shown in a series of runs at 300°C in a 100-A loop that the attack on stainless steel, titanium, and zirconium was dependent upon at least three factors: calcination temperature of the oxide, particle size of the oxide, and the slurry velocity. It was also found that the addition of 1500 ppm of sulfate to the slurry had little or no effect on the attack rates. Slurries prepared from 800°C fired oxide are more suitable than the 650°C fired oxide for use in the various slurry programs insofar as their handling characteristics are concerned. (auth)

9723 HW-26201
[Hanford Works, Richland, Wash.]
CORROSION TESTS–SAE 1010 MILD STEEL IN SYNTHETIC NEUTRALIZED REDOX WASTE SOLUTION. Noborn Endow. Nov. 13, 1952. Decl. Feb. 25, 1957. 5p. Contract [W-31-109-eng-52]. $1.80(ph OTS); $1.80(mf OTS).

9724 K-405
Carbide and Carbon Chemicals Corp. K-25 [Plant], Oak Ridge, Tenn.
CORROSION OF STAINLESS STEELS IN URANYL AMMONIUM PHOSPHATE FILTRATES FROM HANFORD WASTE. H. A. Bernhardt, W. Davis, Jr., and J. R. Flanary. June 1, 1949. Decl. Feb. 23, 1957. 13p. Contract [W-7405-eng-26]. (KLO-103). $3.30(ph OTS); $2.40(mf OTS).

Storage of acid filtrates from the UAP process for uranium recovery from alkaline Hanford waste has imposed the problem of determining the type of metal that can be used for storage tanks. Two stainless steels, 302 and Nb stabilized, have been tested in synthetic filtrates and found to corrode at the rates 0.007 and 0.045 mils/year, respectively. These values are much smaller than an estimated maximum allowable corrosion rate of 10 mils/year. It is concluded that either of these stainless steels or 304 or 316 stainless steels could be used in the construction of tanks for storing acid filtrates from the UAP process. (auth)

9725 NAA-SR-258
North American Aviation, Inc., Downey, Calif.
COMPATIBILITY OF SODIUM, GRAPHITE AND STAINLESS STEEL. T. A. Coultas and R. Cygan. July 8, 1953. Decl. Mar. 5, 1957. 30p. Contract AT-11-1-GEN-8. $0.30(OTS).

Experiments have been performed to determine the compatibility of graphite and molten Na under various conditions. Effects such as corrosion, thermal shock, dimensional changes and penetration of Na into the graphite pores were investigated. Significant corrosive attack on the graphite was noted only in the presence of carburization of the container material. Stainless steel showed no appreciable carburization below 550°C, but carburization was found at temperatures of 600°C and higher. Carburization of Zr was observed only at 750°C and higher. Graphite was found to be wet by liquid Na above 150°C and to absorb the Na into its pores above the level of the bulk liquid by capillary action. Experiments with previously irradiated graphite indicated no significant differences in behavior from that of the unirradiated material. (auth)

Refer also to abstract 9783.

GEOLOGY AND MINERALOGY

9726 FMPC-175
National Lead Co. of Ohio, Cincinnati.
DIGESTION OF URANIUM BEARING ORES USING DILUTE NITRIC ACID. A. B. Kreuzmann and E. R. Johnson. Mar. 4, 1953. Decl. Feb. 16, 1957. 20p. Contract AT(30-1)-1156. $0.30(OTS).

A general equation for the calculation of dilute acid volume and normality required to produce a digest liquor having a specified U concentration and acid normality is developed. Techniques of determining the acid consumption of a feed material are given. Dilute acids are shown to have high efficiency in the digestion of U bearing ores without the formation of appreciable amounts of soluble silica which would interfere with the solvent extraction process. This technique is being amployed at the FMPC Pilot Plant and is proposed for main refinery use. (auth)

METALS AND METALLURGY

9727 ANL-4970
Argonne National Lab., Lemont, Ill.
DECONTAMINATION OF STAINLESS STEEL. F. Johnston and J. J. Katz. Jan. 1953. Decl. Feb. 11, 1957. 14p. Contract W-31-109-eng-38. $0.25(OTS).

The information obtained from absorption curves and radioautographs indicates that the problem of decontaminating stainless steel surfaces which have been immersed

in active UO_2 slurries is one not of removing a particular adsorbed radioactive species but of removing minute UO_2 particles which are tenaciously held by the surface. Consistent with these observations, reagents which effectively dissolve UO_2 or have a corrosive effect on stainless steel were efficient decontaminating agents. Complexing agents, detergents, and wetting agents, on the other hand, were, in general, poor decontaminating agents. An exception was Victamul 116C, a non-ionic organic phosphate produced by the Victor Chemical Co. This product was found to be several times more effective than any of the other mild reagents tested. It did not, however, compare with the oxidizing mixtures in equivalent concentrations. The effectiveness of the $H_2O_2-H_2SO_4$ mixture was markedly enhanced by the addition of Solvadine BL, a wetting agent produced by Ciba. (auth)

9728 ANL-5089
Argonne National Lab., Lemont, Ill.
ZIRCLAD [ZIRCONIUM-CLAD] ZIRCONIUM–URANIUM ALLOY ROD FOR HEAT THRU-PUT TEST. A. B. Shuck. Apr. 17, 1952. Decl. Feb. 12, 1957. 15p. Contract W-31-109-eng-38. $0.25(OTS).

The purpose of this work was to investigate the possibility of producing Zr–clad Zr–U alloy objects by casting the core alloy directly into Zr and Zr–3 wt. % Sn jackets, producing a diffusion bond at the jacket-core interfaces, and to produce small scale heat thru-put test specimens by this method. (auth)

9729 BBC-30
Brush Beryllium Co., Cleveland.
PROGRESS REPORT FOR OCTOBER 1 TO OCTOBER 31, 1948. [SECTION A. FIN TUBE PROJECT. SECTION B. G. E. WASHER PROJECT.] June 10, 1949. Decl. Mar. 25, 1957. 31p. Contract AT-30-GEN-155. $4.80(ph OTS); $2.70(mf OTS).

Three runs were made to produce pressure-tight fin tubes six feet in length. In each of the runs, a newly designed eight-fin envelope was used. The difference between the present envelopes and those used in previous work is found in the shape of the fin section. The early envelopes had a radius at the fin tip; the present envelope has an angular fin tip blending into the walls bounding the fin section. The present design works satisfactorily. A spiral-wound core was centered in an envelope. Be metal powder was charged and compacted by vibrating the envelope during charging. The filled envelope was closed, pressure tested, and placed within the pressure chamber of the fin tube sintering furnace. The envelope was evacuated, then placed under externally applied gas pressure (A gas) of 150 psi. The sintering furnace was slowly (7 hr) heated to the 1050°C sintering temperature and held at temperature for 12 hr. The warm pressing of Be metal blanks to washer shape was completed. The W high-speed steel tools used in the warm pressing proved satisfactory; no serious deformation was encountered. Multiple pressing of washers was tried, in which 2 Be metal blanks were pressed using one double ended plunger and two single ended plungers. The high-speed steel tools were completed and placed in operation. Somewhat over 100 washers were warm pressed from metal blanks. These washers possessed a good detail in the impression and a smooth, even surface. A method wherein metal powder was charged to the warm press die and pressed at 425°C under 100 tsi to form washer compacts, which were subsequently sintered at 1050°C for 3 hr, was also investigated. The compacts possessed well defined form and had a high density (1.835 or over). The sintered washers showed a slight surface oxidation; the oxide scale measured 0.0006 to 0.0009 inch in thickness. (auth)

9730 BBC-50
Brush Beryllium Co., Cleveland.
FABRICATION BY POWDER METALLURGY. Progress Report [for] Period July 1, 1949 through September 30, 1949. C. B. Sawyer, W. W. Beaver, H. W. Dodds, and D. D. Rapprich. May 10, 1950. Decl. Mar. 25, 1957. 33p. Contract AT-(30-1)-510. $4.80(ph OTS); $2.70(mf OTS).

Powder metallurgical techniques are described which were used in the fabrication of special Be shapes. Progress is reported in development of processes for the extrusion of sintered Be; attritioning Be powder; the production of Be–U alloys; and the machining of worm-pressed fluted retainers. Attritioning and hot-pressing techniques for Be are reviewed. (C.H.)

9731 BMI-719(Del.)
Battelle Memorial Inst., Columbus, Ohio.
THE ROLLING OF THORIUM ROD. H. A. Saller, J. R. Keeler, and R. J. Donley. Dec. 19, 1951. Decl. with deletions Feb. 13, 1957. 10p. Contract W-7405-eng-92. $1.80(ph OTS); $1.80(mf OTS).

Thorium rods, $1\frac{7}{16}$ inch in diameter, were rolled from Ames castings. The billets were roughed down in diamond-grooved rolls and finished in hand-rounding rolls. Remelted chips as well as primary-metal castings were rolled, and both yielded satisfactory rods. The surface quality of the rods was generally excellent, and the machined slugs were free of seams. (auth)

9732 BMI-877
Battelle Memorial Inst., Columbus, Ohio.
PROCESSING OF URANIUM–ZIRCONIUM ALLOYS. J. W. Holladay, J. G. Kura, J. H. Jackson, H. A. Saller, R. F. Dickerson, and S. J. Paprocki. Oct. 20, 1953. Decl. Feb. 23, 1957. 22p. Contract W-7405-eng-92. $0.30(OTS).

The preparation and forging and rolling characteristics are reported for the alloys prepared in the CVR program to develop a corrosion-resistant high-uranium alloy. Both arc melting and induction melting were used, but the majority of alloys were prepared by arc melting. Investigated were the binary U-Zr system and ternary U-Zr base alloys with Mo, Nb, Ti, Th, and Sn. Ternary alloys of greatest interest were in the U-Zr-Mo system. (auth)

9733 CF-56-6-87
Oak Ridge National Lab., Tenn.
REPORT OF COPPER DECONTAMINATION WORK AT FEDERATED METALS IN ST. LOUIS. E. S. Bomar. June 14, 1956. 14p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

An industrial-scale experiment involving 100-tons of scrap copper has been completed to check the results of laboratory experiments on the use of fire refining as a means of removing uranium contamination. Samples of the refined metal were checked at ORNL for uranium content by activation analyses and found to have on the average less than 0.66 micrograms of uranium per gram of metal sample. The bulk of the activity was found in the slag. The above level of activity was considered acceptable by the ORO of the AEC and the metal has been released for resale. (auth)

9734 CF-56-6-126
Oak Ridge National Lab., Tenn.
DETECTION OF DENSITY CHANGES IN ThO_2 SLURRY. D. R. Gilfillan. June 21, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Studies were made of γ-ray techniques for detecting density changes in ThO_2 slurry. It is concluded that such a system appears to be feasible. (B.J.H.)

9735 CF-56-8-125
Oak Ridge National Lab., Tenn.
USE OF NITROGEN AS INERT GAS BACKING IN WELDING

AUSTENITIC STAINLESS STEELS. W. J. Leonard. Aug. 20, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

9736 CF-56-11-11
Oak Ridge National Lab., Tenn.
CAPILLARY TUBING FAILURE, IN-PILE LOOP L-2-15. S. E. Bolt. Nov. 1, 1956. 3p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A brief description is given of the failure of the gas addition capillary line in an in-pile loop. A metallographic examination was made of a section of the tubing, and the results are shown. (B.J.H.)

9737 CT-1104
Battelle Memorial Inst., Columbus, Ohio.
PROGRESS REPORT ON METALLURGY OF TUBALLOY TO UNIVERSITY OF CHICAGO. Nov. 10, 1943. Decl. Apr. 1, 1957. 26p. Contract [W-7405-eng-92]. (A-1490). $4.80(ph OTS); $2.70(mf OTS).

Progress is reported on Mg hot dipping of slugs, Pd electroplating of slugs, U-rod cold drawing, U annealing time, analysis of Be alloys and the bonding of Mg to Al by pressure welding and cold rolling. (F.S.)

9738 CT-3714
Iowa State Coll., Ames.
THORIUM METALLURGY. H. A. Wilhelm, A. S. Newton, A. H. Daane, and C. Neher, comps. Feb. 22, 1946. Decl. Feb. 18, 1957. 55p. Contract W-7405-eng-82. $9.30(ph OTS); $3.60(mf OTS).

A comprehensive survey including production, fabrication, physical properties, alloys, and metallography of Th is presented. (T.R.H.)

9739 HW-38453
General Electric Co. [Hanford Atomic Products Operation Richland, Wash.
PREPARATION OF SPHERICAL URANIUM PARTICLES. Report of Invention. William Earl Roake. Aug. 16, 1955. Decl. Feb. 26, 1957. 2p. Contract [W-31-109-Eng-52]. $1.80(ph OTS); $1.80(mf OTS).

A process is described for preparing roughly spherical U metal particles suitable for use in matrix type nuclear reactor fuel elements. The process consists of suspending U or uranium hydride powder in an inert liquid, removing the liquid by sorption, soft-sintering, sintering, and densification. The method is adaptable to continuous processing. (M.P.G.)

9740 ISC-297
Ames Lab., Ames, Iowa.
TERMINAL REPORT OF AN INVESTIGATION OF THE PROPERTIES OF THORIUM AND SOME OF ITS ALLOYS. G. C. Danielson, Glenn Murphy, D. Peterson, and B. A. Rogers. Oct. 1, 1952. Decl. Feb. 26, 1957. 86p. Contract W-7405-eng-82. $0.50(OTS).

Data are included on the physical properties of unalloyed Th, the mechanical properties of production Th, the effect of impurities on the physical properties of Th, the effect of crucible materials on the properties of Th metal, and on the properties of Th alloys and the Th–Zr system. (B.J.H.)

9741 KAPL-1562
Knolls Atomic Power Lab., Schenectady, N. Y.
THE EFFECTS OF POSTIRRADIATION ANNEALING OF URANIUM–ZIRCONIUM ALLOYS. W. V. Johnston. July 5, 1956. Decl. Mar. 28, 1957. 23p. Contract W-31-109-Eng-52. $0.25(OTS).

The possibility of the agglomeration of the inert gases produced during fission and the resultant swelling and blistering of the fuel elements has become a problem of great concern in the design of reactors to operate at highe temperatures and with higher fuel burnups. To obtain som information on the process and the extent of damage to U–Zr alloys, specimens of an 8 wt. % enriched U–Zr alloy which had been irradiated to a maximum of 42% of the U^{235} atoms fissioned were annealed in vacuum furnaces at 425, 510 and 595°C. Densities and the amount of Kr^{85} released were determined as a function of temperature and time up to 500 hr. Specimens of all burnups were stable at 425°C but volume increases of up to 15% were obtained at 595°C. The gas release had a different time dependence at 425°C than at 595°C; activation energies of approximately 11 kcal and 59 kcal, respectively, were obtained. (auth)

9742 KAPL-M-KAK-5
Knolls Atomic Power Lab., Schenectady, N. Y.
PRELIMINARY DESIGN STUDY OF MAGNESIUM–URANIUM ALLOY SLUG. K. A. Kesselring. Feb. 1, 1949. Decl. Feb. 23, 1957. 7p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

9743 KAPL-M-LRM-7
Knolls Atomic Power Lab., Schenectady, N. Y.
THERMAL EXPANSION MEASUREMENTS OF SIX FUEL MATERIALS. Louis R. McCreight. Jan. 10, 1952. Decl. Feb. 23, 1957. 10p. Contract W-31-109-Eng-52. $1.80 (ph OTS); $1.80(mf OTS).

Thermal expansion measurements were made with a Gaertner Interferometer over a low temperature range for UO_2, $MgO-UO_2$, $BeO-UO_2$, $Zr-UO_2$, Zr–U, and Mg–U systems. (T.R.H.)

9744 MITG-A99
Massachusetts Inst. of Tech., Watertown, Mass. Mineral Engineering Lab.
PHOSPHORUS TRACER STUDY OF THE PYROPHOSPHATE PROCESS. James H. Pannell and John J. Brunner. Sept. 22, 1950. Decl. Feb. 26, 1957. 25p. Contract W-7405-eng-85. $4.80(ph OTS); $2.70(mf OTS).

Answers to several questions concerning the fate of excess pyrophosphate used as a selective uranium precipitant in the X or Y cyclic ore-leaching processes were sought in a brief test employing pyrophosphate tagged with P^{32}. It was determined that recycling the barren solution from uranium precipitation was unlikely to cause decreased uranium recovery in the following cycles due to pyrophosphate. (auth)

9745 NYO-3797
Metal Hydrides Inc., Beverly, Mass.
FIFTH QUARTERLY REPORT ON THE REDUCTION OF URANIUM OXIDES. Period Covered: October 1, 1953 through December 31, 1953. Sidney Johnson. Feb. 8, 1954. Decl. Mar. 29, 1957. 16p. Contract AT(30-1)-1350. $0.25(OTS).

The development of evaluation techniques for more precise determination of the purity of UH_3 powders, and their suitability for powder metallurgical processes, was concluded. The evaluation procedures now available are: (1) densities of hot-pressed U pellets processed from UH_3 powder; (2) direct O_2 determinations by the reaction of UH_3 powders with BrF_3; and (3) U oxide assays by dissolution of UH_3 in a measured excess of Ce acid sulfate solution. Utilization of these methods now makes it possible for the first time to evaluate the reductive processes with sufficient accuracy to determine optimum conditions for the variables involved. New furnacing equipment of increased capacity has been placed in operation during this period, and several reduction runs to produce well over one pound of UH_3 have been successfully furnaced. (auth)

9746 NYO-5102
Mallinckrodt Chemical Works, St. Louis.
A REPORT ON THE CASTING OF [SLUGS] IN GRAPHITE MOLDS. William A. Oppold. Apr. 9, 1945. Decl. Mar. 8, 1957. 9p. Contract W-7405-eng-29. $1.80(ph OTS); $1.80(mf OTS).

Casting of satisfactory 1.5-in. diameter and 10.5 in. length U slugs in a five-hole graphite bottom-pour mold is described. Occasional violent boiling was observed, probably caused by improper drying of the mold dressing. Economical production of slugs by this method would require casting in double lengths. (C.W.H.)

9747 ORNL-1647
Oak Ridge National Lab., Tenn.
STATIC LIQUID-METAL CORROSION. Interim Report [Covering] the Period July 1949–September 1952. Anton deS Brasunas. W. D. Manly, ed. May 11, 1954. Decl. Mar. 12, 1957. 99p. Contract W-7405-eng-26. $0.50 (OTS).

Static corrosion tests have been made in molten sodium, lithium, lead, bismuth, and several alloys containing uranium. Temperatures have been varied, but the bulk of the data is concerned with tests made at 1000°C (1832°F). Sodium was found to be the least corrosive in static and dynamic tests, and almost all alloys tested have shown good corrosion resistance. Lithium, on the other hand, attacked many alloys quite severely, especially nickel and nickel-containing alloys. Thermal-gradient dynamic tests have also shown lithium to be much less attractive than sodium because of the susceptibility to mass transfer. Lead and bismuth were quite similar to lithium in many respects, both in static and dynamic corrosion tests, and are compatible with very few metals at 1000°C (1832°F). (auth)

9748 SEP-140
Sylvania Electric Products Inc. Atomic Energy Div., Bayside, N. Y.
SEMI-QUANTITATIVE DETERMINATION OF PREFERRED ORIENTATION IN 600°C ROLLED URANIUM SHEET. Warren Chernock and Donald Wahl. Mar. 17, 1954. Decl. Mar. 7, 1957. 16p. Contract AT(30-1)-GEN-366. (DCF-4709). $0.25(OTS).

A semi-quantitative preferred orientation study was made on uranium sheet rolled at 600°C to 30%, 60%, and 90% R. A. The texture is not well defined after 30% R. A. At 60% R. A. the texture components are (100) [010], (035) [100] as well as the major components of 90% R. A. material. After 90% R. A. the major components are (304) [010], (143) [110], and (223) [140]. A recrystallized 90% R. A. specimen exhibits an increase in the (223) [140] component at the expense of the (143) [110] component. (auth)

9749 SEP-146
Sylvania Electric Products Inc. Atomic Energy Div., Bayside, N. Y.
PRELIMINARY STUDIES OF THE ROLLING OF METAL STRIP FROM METAL POWDER. S. Storchheim, J. L. Zambrow, and H. H. Hausner. Mar. 22, 1954. Decl. Mar. 1, 1957. 58p. Contract AT(30-1)-GEN-366. $0.40(OTS).

Confirmation was made of the fact that powder metals and alloys could be rolled and subsequently sintered into strip form. A number of rolling and powder variables were studied as to their effects upon properties of Al strip. Green Al strip that was hot rolled exhibited high strength and very high density. (auth)

9750 TID-10171
Brush Beryllium Co., Cleveland.
QUARTERLY PROCESS DEVELOPMENT PROGRESS REPORT FOR APRIL, MAY, AND JUNE 1953. A. J. Stonehouse, comp. Oct. 30, 1953. Decl. Mar. 16, 1957. 31p. Contract AT-(30-1)-541. $0.35(OTS).

Current work on the reduction of beryllium fluoride with magnesium and the production of beryllium nitride is described. Studies on the pressing of beryllium powder below and at the sintering temperature are also described. The tensile properties of a number of beryllium-rich powder mixtures containing aluminum or silicon which were fabricated during the current quarter are presented. Quantitative data obtained to date on the effects of the application of vibrational energy to the melting and casting of uranium are summarized. (auth)

9751 WAPD-RM-216
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
THE DEVELOPMENT OF A METHOD FOR CONSUMABLY ARC MELTING CRYSTAL BAR HAFNIUM. J. G. Goodwin and W. J. Hurford. Jan. 8, 1954. Decl. Mar. 7, 1957. 19p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40 (mf OTS).

Three methods are described for melting crystal bar hafnium to produce ingots with the advantages and disadvantages of each listed. The results of the experiment to develop a method for consumably arc melting hafnium crystal bars are summarized as follows. The consumable method developed at WAPD for melting crystal bar hafnium is superior to the non-consumable technique formerly used at WAPD and to the consumable process developed by the Bureau of Mines. The time required to produce a hafnium ingot from crystal bar has been reduced from approximately 6 days to 6 hours through the use of consumable melting. The yield from hafnium crystal bars to conditioned ingots by consumable melting at WAPD is 99.8%. This is slightly greater than the 97% achieved by non-consumable melting and considerably greater than the 78% yield reported by the Bureau of Mines for their consumable melting method. The yield from consumably melted hafnium crystals bars to sheared plates is greater than for the non-consumable type of melt. The yield is now 91%, whereas, it was formerly 85%. In addition, the recycling of scrap raises the over-all yield to 97%. The total number of operations has been substantially decreased. Hot groove rolling, sandblasting and pickling have been eliminated by the consumable melting technique. The possibility of contamination has been decreased by consumably melting. Ingot hardness has been lowered from an average of 184 Bhn/3000 kg for the former non-consumable method to 171 Bhn/3000 kg for the present consumable method. Conditioning of the ingot surface by grinding has been eliminated. Surface welding, the method now employed for conditioning ingots, entails a substantial saving of metal. (auth)

Refer also to abstract 9779.

PHYSICS

9752 ANL-4515
Argonne National Lab., Lemont, Ill.
PHYSICS DIVISION. Quarterly Report for July, August, and September 1950. Oct. 5, 1950. Decl. Mar. 12, 1957. 85p. Contract W-31-109-eng-38. $13.80(ph OTS); $4.80 (mf OTS).

9753 ANL-5040
Argonne National Lab., Lemont, Ill.
REACTIONS PRODUCED BY THE ELECTRICAL EX-

PLOSION OF A METAL IMMERSED IN A FLUID. Robert F. Plott. Dec. 8, 1950. Decl. Feb. 12, 1957. 35p. Contract W-31-109-eng-38. $0.30(OTS).

Calorimetric data are presented from an exploratory experimental study of electrically exploded metal ribbons in water, air, He, and O_2 at several pressures. The metal ribbons so exploded were Al, Au, and U. Electrically controlled exploding equipment made it possible to completely vaporize 20 to 30 mg of Al in air in a blue-white explosion. A two-valve Parr Pressure Bomb and associated calorimeter were used as a means of confining the explosions and measuring the energy exchange. A schematic sketch illustrates the calorimetry of electrically exploded materials. No direct evidence was found of a chemical chain reaction between Al and water during the electrical explosions. (C.H.)

9754 CF-56-7-13
Oak Ridge National Lab., Tenn.
TENTATIVE RECOMMENDATIONS FOR HIGH PRESSURE FLANGE ASSEMBLIES. C. C. Haws, Jr. July 3, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

9755 CF-56-11-2
Oak Ridge National Lab., Tenn.
THERMAL STABILITY OF SODIUM SILICATE TREATED THORIUM OXIDE SLURRY IN DYNAMIC TESTS (100-A LOOP RUN T-79). P. H. Hayes and D. G. Thomas. Nov. 12, 1956. 24p. Contract [W-7405-Eng-26]. $4.80(ph OTS); $2.70(mf OTS).

It was shown that a sodium silicate treated ThO_2 slurry maintained its fluidity during a 100-hour test in a 100-gpm loop at 290°C and 1500 psi. The system was shut down and started up without any difficulty twice during the course of the test. Although the room temperature pH of the slurry was 11.3, the attack rate was less than in many other tests previously made in the 100-gpm loops. The chemical analysis and physical appearance of the samples withdrawn from the loop suggest that the physical chemistry of sodium silicate treated slurry is very complex. (auth)

9756 KT-95
Massachusetts Inst. of Tech., Oak Ridge, Tenn. Engineering Practice School.
BUBBLE FORMATION IN INTERNALLY-HEATED AQUEOUS SOLUTIONS. W. H. Bowman and G. T. Parish. Nov. 15, 1950. Decl. Mar. 1, 1957. 19p. For Carbide and Carbon Chemicals Div. [K-25 Plant]. Contract [W-7405-eng-26, Subcontract 70]. $4.80(ph OTS); $2.70(mf OTS).

A study was made of the transient and steady state phenomena taking place in an internally heated liquid. Complete discussions are included on results and procedures. Photographs of the bubble formation are also included. (B.J.H.)

9757 UCRL-1194
California. Univ., Berkeley. Radiation Lab.
TRANSIENT TEMPERATURES IN FLAT PLATES. Donald N. Hanson. Apr. 6, 1951. Decl. Mar. 7, 1957. 15p. Contract [W-7405-eng-48]. $3.30(ph OTS); $2.40(mf OTS).

Maximum temperature gradients and thermal stresses occur in flat, NaK-cooled, Th target plates when the exterior has cooled faster than the interior, at some point during cooling. The calculations presented here were done to determine the extent of these effects for circular sweeping of the beam. They are easily adaptable to any sweeping pattern. (T.R.H.)

9758 WAPD-RM-222
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
NUMERICAL COMPUTATIONS FOR THE DETERMINATION OF TEMPERATURE DISTRIBUTION IN SPHERICAL PELLETS. Ruth E. Goodman. Feb. 1954. Decl. Mar. 7, 1957. 20p. Contract AT-11-1-gen-14. $3.30 (ph OTS); $2.40(mf OTS).

The analytic method for determining the temperature distribution in and on a spherical pellet in a fixed bed of identical pellets is discussed in detail. The results of machine computations of the temperatures are presented. (B.J.H.)

CRYSTALLOGRAPHY AND CRYSTAL STRUCTURE

9759 A-3814
Carbide and Carbon Chemicals Co. Substitute Alloy Materials Labs., New York.
ORIENTATION OF SINGLE CRYSTALS WITH A UNIVERSAL VISE; GEOMETRICAL THEORY. H. Jacobson. Nov. 19, 1945. Decl. Feb. 15, 1957. 9p. Contract W-7405-Eng-26. (AB-R-46). $1.80(ph OTS); $1.80(mf OTS).

INSTRUMENTS

9760 CF-56-7-128
Oak Ridge National Lab., Tenn.
CALIBRATION OF D_2O RECEIVER (T-2) WEIGH CELL AND THERMAL LEVEL PROBE, HRT-CP EXPERIMENT T-2. J. D. Perret, Jr. July 27, 1956. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The proposed method of liquid-level measurement in the HRT chemical process fuel receiver tank is described. (D.E.B.)

9761 CF-56-8-19
Oak Ridge National Lab., Tenn.
INSTRUMENTATION FOR HRT-CP DISSOLVER SYSTEM. Herman O. Weeren. Aug. 3, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

ISOTOPES

9762 ANL-5077
Argonne National Lab., Lemont, Ill.
PLUTONIUM PRODUCTION ANALYSIS—GRAPHICAL METHOD. J. F. Mumm and L. J. Templin. Jan. 1954. Decl. Feb. 16, 1957. 58p. Contract W-31-109-eng-38. $0.40(OTS).

In the study of plutonium production, the analytical solutions of the various isotope production equations are difficult and time consuming when spatial variation of the flux is considered. In an effort to reduce the time and labor required to determine the integrated production of a given thermal reactor, a graphical method of analysis is presented. The method is based on the assumption that the reactor may be divided into a relatively small number of regions such that the flux in each of these regions may be assumed constant. The production of the given isotope for each region is then read from the appropriate graph. The total production is obtained by adding the productions of all of the regions. (auth)

ISOTOPE SEPARATION

9763 ORNL-1268
Oak Ridge National Lab., Tenn.
SUMMARY OF OPERATING EXPERIENCE WITH DECELERATING ION RECEIVERS. P. E. Wilkinson. Apr. 1,

1952. Decl. Mar. 2, 1957. 21p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

A summary is presented of operating experience with decelerating receivers for ion beam reception. Sample calculations for design of such a receiver for the isotopes of lithium are presented. Estimates of maximum production and requirements to attain such production are given. The maximum ion beam envisioned is 17 amperes obtained from extrapolation of experimental results. (auth)

9764 Y-523
Carbide and Carbon Chemicals Corp. Y-12 Plant, Oak Ridge, Tenn.
CALUTRON ION OUTPUT AND EFFICIENCY AS A FUNCTION OF MAGNETIC FIELD. William B. Cottrell, H. C. McCurdy, and R. S. Livingston. Oct. 1, 1948. Decl. Mar. 6, 1957. 38p. Contract W-7405-eng-26. $6.30(ph OTS); $3.00(mf OTS).

A Beta type calutron has been operated at magnetic fields up to 12,500 oersteds. The field strength necessary for maximum uranium ion output and maximum process efficiency is found to be approximately 8,100 oersteds. Theoretical interpretations are related to the use of uranium tetrachloride charge. (auth)

MEASURING INSTRUMENTS AND TECHNIQUES

9765 WAPD-RM-106
Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.
CALIBRATION OF MARK I FISSION COUNTER AND BF_3 COUNTER. William Baer. Jan. 1952. Decl. Mar. 7, 1957. 15p. Contract AT-11-1-GEN-14. $3.30(ph OTS); $2.40(mf OTS).

Calibration of a Mark I fission counter and a BF_3 proportional counter was performed in the Argonne CP-1 standard pile. The results show that the fission counter sensitivity is 0.70 ± 0.07 counts sec^{-1} $(nv)^{-1}$ for an alpha particle background counting rate of 5.0 counts sec^{-1}. The $B^{10}F_3$ counter sensitivity is found to be 4.44 ± 0.44 counts sec^{-1} $(nv)^{-1}$. (auth)

NEUTRONS

9766 CF-56-5-139
Oak Ridge National Lab., Tenn.
RESONANCE ABSORPTION OF NEUTRONS. CHAPTER 1. SLOWING DOWN OF NEUTRONS IN AN INFINITE HOMOGENEOUS MEDIUM. Lawrence Dresner. June 22, 1956. 26p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70 (mf OTS).

General discussions are included on topics such as neutron slowing down both with and without absorption, the resonance escape probability, and interaction effects. (B.J.H.)

9767 CF-56-9-30
Oak Ridge National Lab., Tenn.
SLOWING-DOWN DENSITY IN A BOUNDED MEDIUM. W. K. Ergen. Sept. 7, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Equations are given for the slowing down density for the case where a spherical shell source, emitting one neutron of age $\tau = 0$ per sec, is embedded in a non-absorbing, homogeneous moderator. (B.J.H.)

9768 NAA-SR-228
North American Aviation, Inc., Downey, Calif.
NEUTRON "TEMPERATURE" INCREASE IN ABSORBERS CAUSED BY SELECTIVE DIFFUSION. E. M. Kelly. Mar. 9, 1953. Decl. Mar. 4, 1957. 28p. Contract AT-11-1-GEN-8. $0.30(OTS).

The "hardening" of neutrons in fuel elements of practical size has been calculated by elementary diffusion theory. It has been assumed that in the D_2O moderator the neutrons have a Maxwellian distribution appropriate to the temperature of the moderator. Within the fuel rod, slowing down of neutrons has been assumed negligible. Graphs are given of relative temperature increase of neutrons in rods, spheres, and slabs as a function of radius. This has been done using three laws of absorption in the fuel element: a v^{-1} law, a $v^{-1.2}$ law, and a combination of these corresponding to natural uranium. Agreement of calculated values with available experimental values is not unreasonable. (auth)

9769 NAA-SR-Memo-35
North American Aviation, Inc., Downey, Calif.
THERMAL DIFFUSION LENGTH IN D_2O. E. R. Cohen. May 29, 1951. Decl. Mar. 27, 1957. 14p. Contract AT-11-1-gen-8. $3.30(ph OTS); $2.40(mf OTS).

The results of measurements on "Lattice 12" have been analysed to determine the thermal diffusion length of the D_2O which is being used in the exponential experiment ("Lattice 12" is actually no lattice at all, merely the exponential tank filled with heavy water). The data give L = 108.1 ± 2.0 cm. Because of the uncertainty in the purity of the heavy water, the diffusion length in 100 per cent D_2O can at present be given only as $L_0 = 163 \pm 48$ cm. (auth)

NUCLEAR PHYSICS

9770 CF-56-7-70
Oak Ridge National Lab., Tenn.
RESONANCE ABSORPTION OF NEUTRONS. CHAPTER 2. DOPPLER EFFECT IN AN INFINITE HOMOGENEOUS MEDIUM. Lawrence Dresner. Aug. 24, 1956. 18p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Equations are developed which indicate the influence of the thermal motion of absorber nuclei on resonance absorption. These equations are used to calculate resonances in U^{238} and Th^{232}. The calculated values are in agreement with experimental measurements. (D.E.B.)

9771 KAPL-1565
Knolls Atomic Power Lab., Schenectady, N. Y.
REPORT OF THE PHYSICS SECTION FOR MARCH, APRIL, MAY 1956. Decl. Mar. 28, 1957. 47p. Contract W-31-109-Eng-52. $0.45(OTS).

Experimental Nuclear Physics. Fission cross section measurements for U^{235} are given in the range 5 to 2000 ev. Fission levels are shown at 8.8, 11.7, 12.3, and 19.3 ev. Semiconducting materials were investigated as positive ion detectors. Initial tests were made on Si and Au-doped Si junctions using 5-Mev α particles. Ion detection by emulsion grain activation was also considered. Statistical methods used in isotopic ratio determinations are discussed. U isotope ratios indicated by the discussed methods are given. Isotopic ratio measurements on U^{234}/U^{236} were complicated by the presence of K_6^{39} ions. Adjustment of the source slit width is shown to solve such complex ion problems. A technique was developed for simultaneously measuring two neighboring isotopes. Using the method, the U^{238}/U^{235} ratio was measured in 0.1 sec. Solid State and Atomic Physics. The study of neutron radiation effects on LiF was extended. F^- formation, embrittlement, and hardness were considered. Preliminary comments on a theoretical study of x-ray scattering effects produced by point defects in crystals are given. Theoreti-

cal Physics. The quasi-homogeneous, long burning reactor was evaluated and found satisfactory. However, a more definite determination of nuclear constants for fuel components in the applicable energy range is necessary. (For preceding period see KAPL-1515.) (D.E.B.)

9772 KAPL-1660
Knolls Atomic Power Lab., Schenectady, N. Y.
PHYSICS REPORT OF THE RESEARCH OPERATION [FOR] SEPTEMBER, OCTOBER, NOVEMBER 1956. Decl. Mar. 13, 1957. 53p. Contract W-31-109-Eng-52. $0.50 (OTS).

Experiments were continued on neutron cross sections of U^{238}. A schematic drawing of the new magnetic α analyzer is shown. The α energy spectrum of Cm^{244} obtained with this spectrometer is given. The operation of the 3-stage mass spectrometer is also discussed. A theoretical study of x-ray scattering by irradiated crystals is included. Studies were also continued on the pulses produced in Si by ionizing particles. Monte Carlo calculations were made of resonance escape in a graphite lattice, and results are shown in graphical form. (For preceding period see KAPL-1611.) (B.J.H.)

9773 UCRL-2150
California. Univ., Berkeley. Radiation Lab.
ENERGY SPECTRUM OF DEUTERONS STRIPPED FROM He^3 AND THE RESULTANT NEUTRON YIELD. F. L. Adelman. Mar. 10, 1953. Decl. Mar. 7, 1957. 6p. $1.80(ph OTS); $1.80(mf OTS).

NUCLEAR PROPERTIES

9774 ANL-4886
Argonne National Lab., Lemont, Ill.
THE THERMAL NEUTRON CROSS SECTION OF 9.2h Xe^{135}. M. S. Freedman. Sept. 12, 1952. Decl. Feb. 12, 1957. 7p. Contract W-31-109-eng-38. This report is a supplement to CP-2782. $1.80(ph OTS); $1.80(mf OTS).

A re-evaluation (cf. report ANL-4030) of the neutron flux measurements used in calculating the thermal-neutron cross section of Xe^{135} given in report CP-2782 indicated that certain important correction factors had not been included and that certain constants were incorrect. A recalculation of the neutron flux values as measured by the relative gold foil activation method and by the U fission-product activation leads to considerably better agreement between the two methods and gives an improved average value for the cross section of $2.55 \pm 0.22 \times 10^{-18}$ cm^2. (G.Y.)

9775 ANL-5150
Argonne National Lab., Lemont, Ill.
A CORRELATION OF FAST NEUTRON FISSION CROSS SECTIONS. John R. Huizenga. Nov. 1953. Decl. Feb. 13, 1957. 7p. Contract W-31-109-Eng-38. $0.20(OTS).

A relationship is shown to exist between the fast (3 Mev) neutron fission cross sections and the nuclear parameter Z^2/A. Some comments are made on the competition between fission and neutron emission. Gamma induced fission results are correlated with neutron induced fission data and the predictions of neutron scattering cross sections from the relative (γ,f) and (γ,n) cross sections are discussed. (auth)

9776 CF-56-6-41
Oak Ridge National Lab., Tenn.
THERMAL NEUTRON CROSS SECTIONS AND DEFINITION OF THE f-FACTOR IN BNL-325. J. A. Harvey. June 7, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The absorption and fission cross sections of U^{233}, U^{235}, and Pu^{239} were measured with monoenergetic neutrons to an accuracy of a few percent in the thermal neutron energy range. The results are tabulated. The definition of the f-factor is also briefly discussed. (B.J.H.)

9777 ORNL-1130
Oak Ridge National Lab., Tenn.
ANALYSIS OF BULK SHIELDING FACILITY NEUTRON DOSIMETER DATA. S. Podger. July 17, 1952. Decl. Mar. 4, 1957. 24p. Contract W-7405-eng-26. $0.25 (OTS).

"Effective removal cross sections" are calculated for Pb, Fe, and O from measurements of fast neutron dose in the water surrounding the BSF reactor. The values for Pb and Fe agree quite well with those previously determined from Lid Tank data, whereas that for O is somewhat lower. (auth)

9778 UCRL-2063
California. Univ., Berkeley. Radiation Lab.
TOTAL NEUTRON YIELD FROM TARGETS BOMBARDED BY DEUTERONS AND PROTONS. Walter E. Crandall and George P. Millburn. Apr. 28, 1953. Decl. Mar. 6, 1957. 40p. Contract W-7405-eng-48. $6.30(ph OTS); $3.00(mf OTS).

The total number of neutrons emitted from various targets bombarded by high-energy deuterons and protons was measured by using a solution of $MnSO_4$ to thermalize and capture the neutrons. The methods of monitoring the number of particles striking the target and of calibrating the neutron detecting apparatus are discussed in detail. Deuterons of energies to 230 Mev and protons of energies to 345 Mev were used. Targets ranging from Li to U were used and were backed by secondary targets of U and Th. The yields are expressed in terms of the number of neutrons emitted per incident particle. U targets gave the highest yield; its yield increases from 2.36 for one range of U (0.625 in.) to 3.25 for 10 in. of U for 190-Mev deuterons. The yield increases with increasing energy of the bombarding particle approximately as the square of the energy in this region. (auth)

NUCLEAR REACTORS

9779 ANL-4617
Argonne National Lab., Lemont, Ill.
DEVELOPMENT OF METHODS FOR CASTING AND FABRICATING ENRICHED URANIUM FUEL SLUGS. Arthur B. Shuck. Apr. 18, 1951. Decl. Feb. 13, 1957. 64p. Contract W-31-109-Eng-38. $0.45(OTS).

Methods of casting and fabricating fuel slugs from one-kilogram reduction buttons were developed on natural uranium. This report describes the equipment and the work done to establish an operational sequence for the manufacture of the fuel for the Experimental Breeder Reactor (CP-4). The purpose of the investigation was to develop methods for producing dimensionally stable fuel slugs from uranium, to obtain a maximum yield of usable slugs from the quantity of uranium available, and to eliminate process losses and to account for the uranium in the process residue. (auth)

9780 ANL-4811
Argonne National Lab., Lemont, Ill.
FISSION RATE MEASUREMENTS IN THE EXPERIMENTAL BREEDER REACTOR (CP-4). Samuel Untermyer. Apr. 18, 1952. Decl. Feb. 12, 1957. 14p. Contract W-31-109-Eng-38. $0.25(OTS).

Fission rates of U^{235}, U^{238} and Th^{232} were determined in CP-4 using a fission ion current chamber. Typical values

obtained are given. These values are in good agreement with measurements at Los Alamos on their fast reactor, and on a fast, tamped critical experiment. This suggests that the amount of extraneous material in CP-4 was not sufficient to degrade the spectrum. It was calculated that the fissions observed in U^{238} would contribute 0.28 to the breeding gain in the reactor. (auth)

9781 ANL-4921
Argonne National Lab., Lemont, Ill.
THE PROPOSED BOILING REACTOR EXPERIMENT. J. R. Dietrich, D. C. Layman, and O. A. Schulze. Nov. 12, 1952. Decl. Feb. 11, 1957. 47p. Contract W-31-109-eng-38. $0.35(OTS).

The stability and safety of boiling reactors will be investigated in a small reactor of variable structure and geometry. Design of the test facility, the experimental program, a cost estimate, and an evaluation of hazards of the experiment are described. Appendices include variables in self-limitation of reactor power, boiling experiments with electrical heating, effect of steam content on reactivity, and a method of estimating power level during equilibrium boiling. (L.T.W.)

9782 ANL-4974
Argonne National Lab., Lemont, Ill.
SUMMARY OF RADIATION MEASUREMENTS RELATING TO THE EXPERIMENTAL BREEDER REACTOR SHIELD. J. W. Butler. Feb. 1, 1953. Decl. Feb. 12, 1957. 16p. Contract W-31-109-eng-38. $0.25(OTS).

A set of selected physical measurements made on the completed Experimental Breeder Reactor (EBR) is presented; the criterion for inclusion was applicability to analysis of shield performance. The information consists principally of fission rate measurements in and near the core and a few photon dose rate observations at the outside of the shield. Also included is a brief statement of plans for future shielding measurements. (auth)

9783 ANL-5079
Argonne National Lab., Lemont, Ill.
DEPOSITION OF CORROSION PRODUCTS BY CATAPHORESIS. Walter Fagan. June 25, 1953. Decl. Feb. 12, 1957. 33p. Contract W-31-109-eng-38. $0.30(OTS).

The prevention of deposition of transport corrosion products on fuel elements and other critical components of a circulating water nuclear reactor was investigated. As the deposition observed was similar to that in which Fe filings are used to observe the magnetic field of a horseshoe magnet, cataphoretic removal of the corrosion products was suggested. The high-pressure cataphoresis pipe test section is described. The results showed that such a deposition was possible. The history of a charged particle in an electric field subject to a turbulent-type resisting drag is calculated. (J.S.R.)

9784 ANL-5095
Argonne National Lab., Lemont, Ill.
DETERMINATION OF THE CONVERSION RATIO OF THE EXPERIMENTAL BREEDER REACTOR BY RADIOCHEMICAL METHODS. "PRODUCTION AND CONSUMPTION PATTERNS IN THE EBR." M. Levenson. Dec. 23, 1953. Decl. Feb. 12, 1957. 63p. Contract W-31-109-Eng-38. $0.40(OTS).

An evaluation of breeding in the EBR by chemical methods is described. The analytical methods and results for determining Pu production patterns and U consumption patterns involved the accurate measurement of neutron spectra, cross sections for fission and capture, and resulting values of α for U^{235} and Pu^{239} These procedures are outlined in detail. A net conversion ratio of 1.00 was obtained. Of the total Pu produced, 63.9% occurred in the outer blanket and 35.7% in the inner blanket. (K.S.)

9785 ANL-5189
Argonne National Lab., Lemont, Ill.
PRESSURE DROP TESTS ON TWISTED RIBBON CORE ASSEMBLIES. A. B. Schultz. Aug. 1953. Decl. Feb. 11, 1957. 42p. Contract W-31-109-eng-38. $7.80(ph OTS); $3.30(mf OTS).

Twisted ribbon fuel elements are a completely new concept for reactor core construction. Advantages of this construction are outlined briefly, together with a list of ribbon sizes and core designs now under consideration. A test program was conducted to determine pressure drop through core assemblies of various geometries. Cold water and water-air mixtures were used to simulate two-phase flow conditions. The results are presented in tabular and graphical form. Attempts were made to correlate experimentally determined pressure drops with theoretical data. The pressure drop through twisted ribbon cores appears to be much less than originally anticipated and the empirical data presented permits reasonably accurate calculations of any proposed design. (auth)

9786 ANL-5222
Argonne National Lab., Lemont, Ill.
A PHYSICAL DETERMINATION OF THE CONVERSION RATIO OF THE EXPERIMENTAL BREEDER REACTOR. C. D. Curtis, S. H. Klein, D. Okrent, W. C. Redman, and S. Untermyer. Aug. 1954. Decl. Feb. 12, 1957. 114p. Contract W-31-109-eng-38. $0.55(OTS).

Physical methods of measurement have been employed to determine the ratio of Pu^{239} production to U^{235} destruction in the Experimental Breeder Reactor (CP-4). Highly depleted U^{238} foils were irradiated throughout the core, inner blanket and outer blanket (cup), and the U^{238} neutron capture rates were determined by scintillation counting of the plutonium $K\alpha_1$ and $K\alpha_2$ x rays which accompany the decay of Np^{239} The observed activities required corrections for contributions by fission products and the decay of U^{237} formed by an (n,2n) threshold reaction of U^{238}. A U^{235}-coated fission chamber of the gamma-ray compensated type was used to establish the fission patterns in the core and inner blanket. These measurements were supplemented by the irradiation and counting of copper and U^{235} foils placed in those portions of the outer blanket which were inaccessible to the current chamber. The observations for Pu^{239} production and U^{235} fission were normalized by a comparison in the EBR thermal column of the response of the detectors to a thermal neutron flux for which pertinent cross sections are known. When combined with values for the ratio of captures to fissions in U^{235}, as determined by mass spectrometric analysis of fuel from the EBR, the measurements yielded a value for the conversion ratio which is in excellent agreement with that determined by radiochemical procedures. It was found that 91% of the U^{235} destruction occurs in the core, 3% in the inner blanket, and 6% in the outer blanket. One-half per cent of the Pu^{239} produced is found in the core, 33-1/2% in the inner blanket and 66% in the outer blanket. For the movable outer blanket in its position of maximum effectiveness, the physically determined conversion ratio is 1.01 ± 0.05. Estimates based on these measurements indicate that a conversion ratio of 1.3 is possible for a fast converter of the EBR type. (auth)

9787 ANL-5283
Argonne National Lab., Lemont, Ill.
THE REGENERATION FACTOR AS A FUNCTION OF TIME IN A Th^{232}–U^{235} THERMAL REACTOR. J. C. Carter. Sept. 1954. Decl. Feb. 12, 1957. 24p. Contract W-31-109-Eng-38. $0.25(OTS).

A theoretical investigation of the variation of the regeneration factor η in a Th^{232}-U^{235} thermal reactor was made. The abundances of the significant isotopes in the thorium-

uranium cycle have been derived as a function of irradiation time at constant reactor power. The change in η as a function of irradiation time at constant power was calculated for combinations of enrichment and resonance escape probability considered likely to exist in a thermal reactor. The effect upon η of the absorption cross section of $_{91}Pa^{233}$ and of the fission products has been shown. (auth)

9788 ANL-LB-SL-1053
Argonne National Lab., Lemont, Ill.
EFFECT OF PROCESSING LOSSES ON BREEDING GAIN AND DOUBLING (CONVERTING) TIME FOR EBR-II AND PBR REACTORS. I. G. Dillon. May 8, 1956. Decl. Mar. 9, 1957. 4p. Contract [W-31-109-eng-38]. $1.80(ph OTS); $1.80(mf OTS).

9789 BNL-423
Brookhaven National Lab., Upton, N. Y.
REACTOR OPERATIONS DIVISION MONTHLY REPORT [FOR] OCTOBER 1956. R. W. Powell. Oct. 1956. Decl. Feb. 13, 1957. 19p. $0.30(OTS).

A summary of reactor operations and related activities is given. (For preceding period see BNL-420.) (F.S.)

9790 BNL-2787
Brookhaven National Lab., Upton, N. Y.
CRITICAL EXPERIMENTS FOR THE BROOKHAVEN MEDICAL REACTOR: SUMMARY HAZARDS REPORT. June 10, 1955. Decl. Feb. 12, 1957. 45p. $0.35(OTS).

The hazards associated with a series of critical experiments planned to be carried out in the Brookhaven critical experiment facility as an aid in the design of the Brookhaven Medical Reactor are discussed. The critical assembly and its operation are described, general hazards are outlined, radiation doses from a runaway are calculated, and steps to minimize the risk are outlined. (C.H.)

9791 CF-53-10-185
Oak Ridge National Lab., Tenn.
COOLING TOWER REACTOR. J. P. Sanders and P. N. Haubenreich. Oct. 14, 1953. Decl. Mar. 21, 1957. 21p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

A preliminary design has been made of a homogeneous reactor in which the heat is removed from the core solution in a packed or slatted cooling tower. In this tower a portion of the moderator is evaporated into a recirculating gas stream. This gas is cooled and a portion of the vapor is condensed in a heat exchanger producing steam. The operating conditions of the cooling tower chosen for a 50-Mw reactor include a gas flow rate of 268,820 pounds of oxygen per hour and a total pressure of 700 psia. The inlet and outlet gas temperatures are 390 and 455°F, respectively. The liquid enters the tower at 482°F and leaves at 416°F. The required volume of packed tower, assuming a transfer coefficient of 600 Btu/hr·ft^2·enthalpy difference, is 1200 cubic feet. A flow sheet outlining the operating conditions in this reactor is shown. (auth)

9792 CF-56-5-124
Oak Ridge National Lab., Tenn.
CONTROL AND START-UP OF HRR. Paul R. Kasten. May 18, 1956. 3p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

9793 CF-56-5-143
Oak Ridge National Lab., Tenn.
HRR HEAT EXCHANGERS. Howard A. McLain. May 21, 1956. 5p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A study has been made to determine the approximate size of heat exchangers required for heat removal from the homogeneous research reactor. Although limited time prevented the selection of an optimum size heat exchanger, sufficient calculations were made to select one near the optimum size. Six 83.3-Mw heat exchangers are required to remove 500 Mw of heat from the reactor. Calculations show that the requirements for these heat exchangers is very similar to the blanket heat exchanger for the TBR. On the basis of the TBR studies, a two drum steam generator is selected and its specifications are listed. (auth)

9794 CF-56-5-190
Oak Ridge National Lab., Tenn.
ORR PROJECT PROGRESS REPORT NO. 18. J. P. Gill and T. E. Cole. May 15, 1956. 18p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Progress is briefly reported on the design and construction of the ORNL Research Reactor. Experimental facilities are described. Proposed experiments for the reactor are listed. (For preceding period see ORNL-CF-56-2-158.) (B.J.H.)

9795 CF-56-6-28
Oak Ridge National Lab., Tenn.
HRR CONVERTER PLATES. D. R. Gilfillan and H. A. McLain. June 1, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A converter suitable for use inside a 3-in. diameter thimble of a homogeneous research reactor has been proposed for obtaining high fast neutron fluxes. The limitation on the amount of fast neutron flux that can be obtained from such a converter is dependent on the rate that the generated heat can be removed from this unit. A converter unit consisting of a series of thin concentric cylinders shows much promise of producing fast neutron fluxes of the order of 1 to 5×10^{15} neutrons/(sq cm)(sec). This fast flux refers to the number of unmoderated neutrons passing through a 1 sq cm area normal to the neutron velocity at the geometrical center of the cylinders. The proposed converter consists of seven concentric cylinders 2 or 4 in. long, 1 mm thick, with 3 mm spacing between cylinders. The radius of the inner cylinder is 0.48 in. and that of the outer cylinder is 1.42 in. Larger cylinder lengths do not appear desirable because of the attenuation of the fast neutrons by the cooling water. Thinner cylinder plates would permit higher neutron fluxes and efforts should be made to reduce the thickness of these plates to the practical minimum. (auth)

9796 CF-56-6-29
Oak Ridge National Lab., Tenn.
ECONOMICS OF POWER GENERATION—HOMOGENEOUS RESEARCH REACTOR. M. I. Lundin. June 4, 1956. 6p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Preliminary investigations were made of the economics of generating power from a homogeneous research reactor. Tables are given which show the variation of installed and yearly charges for a turbo-generator facility, estimated costs for a heat rejection system based on the use of air-cooled steam condensers, and the yearly power revenue as a function of station load factor and power charges. (B.J.H.)

9797 CF-56-6-80
Oak Ridge National Lab., Tenn.
RATE OF HEAT REMOVAL FROM FUEL REFLUX CONDENSER. HRT Report II A 15B. Work Period: June 8,

1956. P. N. Haubenreich. June 13, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A test has been carried out to determine rates of heat removal from the fuel reflux condenser under various conditions. The maximum rate at which heat could be removed by the condenser cooling water was measured at a number of different pressures in the storage tank and reflux condenser. Results indicate satisfactory performance. (auth)

9798 CF-56-6-123
Oak Ridge National Lab., Tenn.
HRT CRITICAL CONCENTRATION AS A FUNCTION OF TEMPERATURE. C. W. Nestor, Jr. June 20, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Two-group, two-region calculations were made of the HRT critical concentration, and results are shown graphically. (B.J.H.)

9799 CF-56-6-137
Oak Ridge National Lab., Tenn.
RATE OF HEAT REMOVAL FROM BLANKET REFLUX CONDENSER. HRT Report IV A 39b. Work Period: June 15, 1956. P. N. Haubenreich and H. E. Williamson. June 21, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A test has been carried out to determine rates of heat removal from the blanket reflux condenser under various conditions. The maximum rate at which heat could be removed by the condenser cooling water was measured at a number of different pressures in the storage tank and reflux condenser. Results are shown. (auth)

9800 CF-56-6-171
Oak Ridge National Lab., Tenn.
HRT COOLING TOWER PERFORMANCE TEST EVALUATION. R. E. Brooksbank. June 28, 1956. 5p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The HRT cooling tower has been tested and was found to exceed design performance by approximately 20%. (auth)

9801 CF-56-7-38
Oak Ridge National Lab., Tenn.
MAINTAINING HIGH PRESSURE SYSTEM OF THE HRR AT TEMPERATURE AFTER DUMP. C. Michelson. July 10, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

It has been suggested by Kasten and others that the HRR be shut down by partially dumping the fuel solution and allowing the remaining fission product decay heat to keep the high pressure system at temperature during shutdown. An investigation of the technical feasibility of this suggestion is the subject of this report. The results of this preliminary study indicate that such an idea is feasible if the high pressure system is kept at an equilibrium temperature in the range 350 to 390°F during shutdown. A fuel solution holdup of 9,600 to 11,900 liters (7.7 to 9.5 kg) is required. A minimum of 9.5 Kg will make the reactor go critical at 390°F; about 8.7 Kg at 350°F. Most of the fuel solution will be in the return pipes to the reactor. The possibility of an incident which will transfer fuel to the reactor and establish a new critical geometry with a smaller fluid and fissionable material inventory is extremely remote. A detailed nuclear study if required, however, to select the safest equilibrium temperature. For initial design of the HRR, an equilibrium temperature of 190°C (374°F) is selected. A fuel holdup of about 11,000 liters (8.8 Kg) is required (critical mass, 9.3 Kg). The maximum heat demand on the system two days after shutdown is about 240 Kw; 207 Kw direct heat losses and 33 Kw evaporation requirements for gas dilution. A D_2O make up of 6800 liters is required during a two day shutdown to maintain constant fuel concentration. With a gas dilution to 5 mole percent D_2, and subsequent venting, there should be no explosion hazard. Removal of heat in excess of the requirements for venting and heat losses can be accomplished by controlled boiling at a lower pressure in one or more of the heat exchangers. (auth)

9802 CF-56-7-58
Oak Ridge National Lab., Tenn.
FUEL CHARGING SYSTEM. HRT-CP Engineering Test T-11 and T-16. R. E. Brooksbank. July 12, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

9803 CF-56-7-62
Oak Ridge National Lab., Tenn.
PROBLEM OF GAS RECOMBINATION CAUSED BY INITIAL OPERATION AT REDUCED TEMPERATURE. Paul N. Haubenreich. July 17, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The gas recombination problems in the HRT caused by initial operation at reduced temperature are briefly discussed. The concentrations of copper required in the fuel solution for various conditions are tabulated. Solutions to the gas recombination problems are suggested. (B.J.H.)

9804 CF-56-7-63
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTORS. Design Section Progress Report for June 1956. W. R. Gall and M. I. Lundin. July 13, 1956. 14p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

Progress is reported on the development of the cooling water system, refrigeration system, fuel storage tanks, corrosion specimen holders, and flanges for the HRT. A feasibility study was made on a circulating fuel aqueous homogeneous research reactor. The basic design parameters are included. (B.J.H.)

9805 CF-56-7-71
Oak Ridge National Lab., Tenn.
RESONANCE ABSORPTION OF NEUTRONS. CHAPTER 3. RESONANCE ABSORPTION IN HETEROGENEOUS MEDIA. Lawrence Dresner. Aug. 10, 1956. 15p. Contract W-7405-eng-26. $3.30(ph OTS); $2.40(mf OTS).

Equations for the calculation of neutron resonance absorption in heterogeneous matrices are presented. (W.L.H.)

9806 CF-56-7-87
Oak Ridge National Lab., Tenn.
ESTIMATES OF TEMPERATURE RISE AND PRESSURE DROP EFFECTS OF SAMPLE HOLDER IN THE HRT CORE. R. E. Aven. July 17, 1956. 9p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A study has been made to determine the surface temperature of structural components of the sample holder which is to be placed in the core of the HRT and to see if boiling might occur around the holder. Three structural materials, titanium, Zircaloy and stainless steel, have been considered. Heat generation rates in these metals have been estimated for operation at 5 Mw and at 10 Mw and are listed. By estimating the surface to volume ratio of various components of the holder it is shown that the higher surface temperature will probably occur in the structural Y-beam. Assuming natural circulation it is possible to estimate the temperature drop between the Y-beam surface and the main bulk of the fuel solution. The temperature drop required to remove the heat generated in a titanium or Zircaloy Y-beam is ~ 25°F (5 Mw) while that required for a stainless steel Y-beam is ~ 35°F. The calculations indicate that if titanium or zircaloy is used the surface temperature will be ~ 314°C (max) and no boiling of the fuel solution will occur unless the total pressure falls below ~ 1540 psia. If stainless steel is used the fuel solution around the holder may boil if the pressure is

lowered to 1650 psia. Either titanium or Zircaloy is recommended as the structural material for the sample holder. Care should also be taken to remove any pockets or crevices in the holder where fuel might be trapped and become stagnant. Several batteries of samples will be placed in the exit line from the core vessel. The greatest restriction to flow occurs at the ring holder assembly. An estimate of the pressure drop at this point for normal operation has been made and is ~ 0.7 psi. If four sample batteries are used in the exit line the additional pressure drop will be ~ 3 psi. (auth)

9807 CF-56-7-107
Oak Ridge National Lab., Tenn.
SHUTDOWN AND STARTUP PROCEDURE FOR HRR. C. Michelson. July 24, 1956. 13p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

A fast startup procedure for the homogeneous research reactor is suggested. The proposed procedure is based on the retention of sufficient fuel in the high-pressure system at shutdown to maintain a system temperature of 374°F. Criticality would then be attained at 482°F by the addition of preheated D_2O and concentrated fuel solution. This procedure would required 70 min. and greatly reduce thermal stresses. (D.E.B.)

9808 CF-56-7-116
Oak Ridge National Lab., Tenn.
RESULTS OF HRT TEST I B 5a, VOLUME CALIBRATION OF FUEL HIGH-PRESSURE SYSTEM. C. C. Cardwell and P. N. Haubenreich. July 30, 1956. 4p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

9809 CF-56-8-49
Oak Ridge National Lab., Tenn.
EFFECT OF XENON-135 RETENTION UPON HOMOGENEOUS REACTOR BEHAVIOR. Melvin Tobias. Aug. 9, 1956. 20p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

To investigate the effects of Xe concentration transients, calculations were performed for the case of a 4-ft diameter spherical reactor operating at 280°C, containing U^{235}, D_2O, I^{135}, and Xe^{135}, and at initial fluxes of 10^{13} to 10^{15} n/cm^2-sec. The reactor power was reduced to fractions of either 0 or 0.1 of the initial value, and when the Xe concentration had reached a maximum, the reactor power was increased to its original value. The results obtained show that rates of concentration change, temperature rise, and reactivity addition which may be associated with the build-up or burn-out of Xe are low enough to allow sufficient time for corrective measures. (auth)

9810 CF-56-8-124
Oak Ridge National Lab., Tenn.
TRANSIENT CONDUCTION IN NUCLEAR REACTOR FUEL PLATES. Lloyd G. Alexander. Aug. 20, 1956. 27p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

The problem of transient conduction in an unclad fuel plate with negligible surface resistance was solved analytically for the case wherein simultaneous step changes are made in the volume distributed heat generation rate and the surface temperatures. The solutions for the central fuel temperature and the space-mean temperature are presented in graphs. A solution for the case of an arbitrary number of successive step changes was obtained by superposition, and an approximate solution for the case of arbitrarily and continuously varying heat release rate and surface temperatures coupled to the reactivity was discussed. (auth)

9811 CF-56-8-134
Oak Ridge National Lab., Tenn.
HRT DRAWINGS—CROSS REFERENCE LISTING. F. C. Zapf. Aug. 13, 1956. 32p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

9812 CF-56-8-152
Oak Ridge National Lab., Tenn.
HRT COOLING WATER SYSTEM MODIFICATIONS. H. A. McLain, R. E. Aven, and R. C. Robertson. Aug. 23, 1956. 37p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

A design study of the HRT cooling water system is summarized with regard to a rearrangement to provide a closed circulating loop of demineralized water for the stainless steel cooling equipment located within the reactor cell. The demineralized water system acts as a liquid couple for removing heat from the reactor cell to the cooling tower water. The maximum steady state load on this system is 3,480,000 Btu's per hour. A shell and tube heat exchanger with 800 sq. ft. of effective surface area, a 300 gpm (120 ft. head) circulating pump, and miscellaneous smaller equipment are required to make the necessary changes. The total installed cost should range between $30,000 and $40,000. (auth)

9813 CF-56-8-175
Oak Ridge National Lab., Tenn.
FLOW CHARACTERISTICS FOR HRT PRESSURE-BALANCING CONTROL VALVE. Carlyle Michelson. Aug. 28, 1956. 16p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

This report presents the basic criteria and calculations leading to the specification of a pressure-balancing control valve for the HRT. Included is a suggested layout for connecting the valve into the present piping system. (auth)

9814 CF-56-8-189
Oak Ridge National Lab., Tenn.
DESIGN SECTION MONTHLY REPORT [FOR] JULY AND AUGUST 1956. W. R. Gall and M. I. Lundin. Aug. 29, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Design studies on various HRT components are briefly summarized. The normal operating conditions of the modified HRT cooling water system are tabulated. (B.J.H.)

9815 CF-56-9-2
Oak Ridge National Lab., Tenn.
TIME REQUIRED FOR VENTING THE HRT PRESSURIZER DURING A DUMP. L. C. Wilbur. Sept. 4, 1956. 11p. Contract [W-7405-Eng-26]. $3.30(ph OTS); $2.40(mf OTS).

Calculations were made which indicate that the venting process using the present valve trim will require from 9 to 13 minutes. The valve trim should be modified to permit an increased flow rate if a 1 to 2 min venting time is desired. Curves are given of the proposed flow rate and the associated rate of system pressure decrease. (B.J.H.)

9816 CF-56-9-24
Oak Ridge National Lab., Tenn.
HRP-CP: AN ANALYSIS OF THE GAS SEPARATOR FOR USE AS A SOLIDS CONCENTRATOR. W. L. Carter and F. C. McCullough. Sept. 4, 1956. 11p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

The gas separator, which is in the circulating fuel stream of the HRT, should provide appreciable concentration of fission and corrosion product solids along the wall surface of the unit. A considerable centrifugal field is induced by the swirling motion imparted to the flowing stream, and the calculations indicate that particles of 8.7-micron diameter and larger will all be thrown to the wall. Smaller particles will be concentrated also but to less degree. A plot of particle size versus percent of particles separated from the solution is included. This study indicates that a feed stream to the core processing plant taken from the outer periphery

of the gas separator would contain a higher concentration of solids than a stream tapped from a pipe line. However, to provide appreciable advantage for solids collection and removal from the walls of the gas separator, considerable redesign of the unit would be necessary. In view of the delay to HRT construction that would be brought about by redesigning the gas separator, it is not advisable to recommend changing the feed point of the core processing plant. (auth)

9817 CF-56-9-35
Oak Ridge National Lab., Tenn.
RADIATION PROBLEM IN REMOVING RUPTURE DISC ASSEMBLY AFTER OPERATION OF HRT. R. E. Aven. Sept. 10, 1956. 10p. Contract [W-7405-eng-26]. $3.30 (ph OTS); $2.40(mf OTS).

An estimate has been made of the radiation involved in removing a rupture disc assembly after the reactor has operated for a period of time. Two cases were considered: where the only radiation involved is the induced radioactivity in the metals; and radiation includes a film of fuel solution coating the inside of the box assembly housing the rupture discs. The estimates show that the radiation from the fission products in the rupture disc assembly may be as much as 300 times as great as the neutron induced activity in the metal. The greatest radiation hazard occurs when the rupture disc assembly, coated with fuel solution, is removed from the biological shield. It is planned at present to pull the rupture discs into a lead container with about four inches of lead on the top and sides, but with the bottom open. To reduce radiation from gamma scattering to a safe level, even at a distance of 20 to 30 feet, the rupture discs should be pulled into the lead shield so as to be at least 9 inches from the bottom opening. It may be better to provide a lead tray or plug to cover this opening in the rupture disc container. (auth)

9818 CF-56-10-43
Oak Ridge National Lab., Tenn.
HRT DUMP TEST, PHASE II. C. H. Gabbard. Oct. 4, 1956. 8p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The control system for maintaining a balance of the core and blanket pressures during a dump of the HRT was evaluated, and the corrections necessary to maintain a pressure balance of ±50 psi were made. A schematic diagram of the system is given. Results are graphed. (M.H.R.)

9819 CF-56-10-50
Oak Ridge National Lab., Tenn.
EFFECT OF GAS SAMPLING ON THE HRT LEVEL CONTROL. R. F. Hughes. Oct. 15, 1956. 4p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A mathematical analysis of the problem has been made. The conclusion reached is that the vent line is sufficiently large to preclude a difference in reactor and indicator fluid levels during gas sampling. (M.H.R.)

9820 CF-56-10-53
Oak Ridge National Lab., Tenn.
NATURAL CIRCULATION HOMOGENEOUS REACTORS. M. W. Rosenthal. Oct. 16, 1956. 23p. Contract [W-7405-eng-26]. $4.80(ph OTS); $2.70(mf OTS).

Elimination of the fuel circulating pump from a homogeneous reactor dispenses with a major potential source of operating difficulty. Computations reveal that power levels achievable by natural circulation (without phase change) are of the order of $\frac{1}{5}$ to $\frac{1}{3}$ of that obtainable in a forced flow system using the same core and with the same heat-exchange area. Nautral circulation power is a function of the height of the system, the flow resistance, and the temperature limitations on the fuel solution. The operating dependability of natural circulation may recommend it for applications in which high power density machines are not required and maintenance-free operation is necessary. In particular, natural circulation would be worth consideration for a solution system if power density were already limited by corrosion rate. (auth)

9821 CF-56-10-129
Oak Ridge National Lab., Tenn.
INDUCED ACTIVITY IN HRT CORE SPECIMEN HOLDERS. E. H. Gift. Oct. 30, 1956. 12p. Contract [W-7405-eng-26]. $3.30(ph OTS); $2.40(mf OTS).

An estimate has been made of the expected neutron induced activation in the metallurgical specimens and specimen holder which are to be inserted into the core and blanket regions of the HRT. The activity of the fission products which may be adhering to the specimen holder has also been estimated. These estimates indicate that for estimated thermal neutron fluxes of 2×10^{14} and exposure times of one year, the maximum induced activity one hour after shutdown will be approximately 600 r/hr, and the adhering fission product activity will be approximately 61.2 r/hr at a distance of 6 ft in air. It was then calculated that 9.45 inches of lead were required to reduce this dose rate to an acceptable one of 7.5 mr/hr at the outside surface of the lead. (auth)

9822 CF-56-11-25
Oak Ridge National Lab., Tenn.
INSTRUMENTATION FOR HRT BLANKET PROCESSING SYSTEM. Herman O. Weeren. Nov. 5, 1956. 5p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The instruments which are known to be required for the HRT blanket processing system are listed. Available information on service conditions is also included. (B.J.H.)

9823 CF-56-11-82
Oak Ridge National Lab., Tenn.
THE INTEGRAL SPECTRUM METHOD FOR GAMMA HEATING CALCULATIONS IN NUCLEAR REACTORS. L. G. Alexander. Nov. 21, 1956. 39p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

Examples are presented that illustrate the application of the straight-ahead scattering, integral-spectrum method to the estimation of γ heating in nuclear reactors for cases of relatively simple geometry and source distribution. Equations are included for the calculations of γ heating for more complex geometries than those listed above. (W.L.H.)

9824 CF-56-11-88
Oak Ridge National Lab., Tenn.
HRP-CP: PIPE LINE DESIGNATION TABLE FOR THE HRT CHEMICAL PROCESSING PLANT–REVISION 1. H. O. Weeren. Nov. 21, 1956. 35p. Contract [W-7405-eng-26]. $6.30(ph OTS); $3.00(mf OTS).

9825 CF-56-11-112
Oak Ridge National Lab., Tenn.
FAILURE OF LET-DOWN VALVE IN SLURRY BLANKET TEST. L. F. Parsly. Nov. 15, 1956. 8p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

The slurry blanket test let-down valve failed after 50 cycles and a total flow time of about 27 minutes in 500 g Th/kg H_2O slurry. The valve is a conventional control valve with stellite trim. The failure was due to erosion of a limited area of the seat and poppet. (auth)

9826 CF-56-11-126
Oak Ridge National Lab., Tenn.
DESIGN SECTION MONTHLY REPORT [FOR] NOVEMBER 1956. W. R. Gall and M. I. Lundin. Nov. 27, 1956. 7p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Modifications in the HRT cooling water system are

briefly discussed. Discussions are also included on the refrigeration system, water treatment, blast shields, and other phases of the HRT. (B.J.H.)

9827 CF-56-11-131
Oak Ridge National Lab., Tenn.
1. SUMMARY OF RUNS, S56, 57, 73, 74, 78, 79, 80, 87, 88, 89 IN WHICH SLURRIES CIRCULATED AT 300°C IN 100-A LOOPS CAKED OR ACCUMULATED ON PIPE WALLS. 2. SUMMARY OF RUNS T-52 THROUGH T-75 IN WHICH A BYPASS LINE WAS USED IN PARALLEL WITH MAIN CIRCULATING LOOP. A. S. Kitzes, R. B. Gallaher, C. A. McLaughlin, and R. P. Wichner. Nov. 16, 1956. 39p. Contract [W-7405-Eng-26]. $6.30(ph OTS); $3.00(mf OTS).

9828 CF-56-12-41
Oak Ridge National Lab., Tenn.
A MANOMETER SYSTEM FOR A HIGH TEMPERATURE SLURRY LOOP: REPORT OF RUN S-91. R. P. Wichner and A. S. Kitzes. Dec. 7, 1956. 7p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A manometer line purge system utilizing pressurizer condensate was linked with the "S"-loop venturi prior to run S-91. A differential pressure (D/p) cell, connected across the purge lines, measured the slurry flow rate throughout the run which lasted 72 hours. The system worked well and inspection of the manometer lines after termination of the run showed them to be free of slurry. (auth)

9829 DP-32
Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.
INITIAL OPERATION OF THE STANDARD PILE. R. C. Axtmann, L. A. Heinrich, R. C. Robinson, O. A. Towler, and J. W. Wade. Dec. 1953. Decl. Feb. 15, 1957. 43p. Contract AT(07-2)-1. $0.35(OTS).

The Standard Pile was first brought to criticality on July 25, 1953. The first two months of operation permitted calibration of the reactor and preparation of it for use as a neutron source in exponential experiments. (auth)

9830 DP-133(Del.)
Du Pont de Nemours (E. I.) & Co. Savannah River Lab., Augusta, Ga.
STRESS ANALYSIS OF THE SRP REACTOR TANK. J. W. Walker. Oct. 1955. Decl. with deletions Feb. 18, 1957. 24p. Contract AT(07-2)-1. $4.80(ph OTS); $2.70(mf OTS).

Equations are derived for the discontinuity stress in a cylindrical tank that is mounted vertically on a header and is open at the top. The analysis includes a derivation of equations for the prediction of the deflection and slope of the header by relating the strain energy in bending a circular plate to the strain energy in bending the header. The stresses that result from pressure and temperature are included in the analysis. (auth)

9831 HW-24282
Hanford Works, Richland, Wash.
SMALL SOURCE MODEL OF A THERMAL PILE. W. A. Horning. Apr. 30, 1952. Decl. Feb. 26, 1957. 10p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

For many problems of the type in which the total number of cells in the pile is small, and in which a small number of cells in the pile differs markedly from the other cells, or in which one wishes to know the effect on the pile of changes in the size of the rods or in their spacing, a small source approximation seems appropriate. In this approximation the neutron sources are assumed concentrated in points, lines, or surfaces whose spacing equals the spacing of the actual sources of finite volume. The slowing of the neutrons may be treated on the one-group, two-group, or age theory method when the small-source model is used. The motion of thermal neutrons is treated by ordinary diffusion theory. The approximation is applied to various pile problems. (G.Y.)

9832 KAPL-100
Knolls Atomic Power Lab., Schenectady, N. Y.
A GRAPHICAL SOLUTION FOR THERMAL AND PARTIALLY THERMAL PILES. H. E. Stevens, Jr. and W. G. Gibbard. Dec. 27, 1948. Decl. Feb. 26, 1957. 33p. Contract W-31-109-Eng-52. $0.30(OTS).

A graphical method of making two-group critical mass calculations for homogeneous thermal piles is presented, using a modified two-group theory in which the effects of epithermal fissions and absorptions are included. The analysis leading to the graphical solution was carried out for spherical, reflector covered piles. In addition to the method presented considerable data obtained in this manner for Be moderated, pure U^{235} piles with Na coolant have been included. (auth)

9833 KAPL-M-JZ-1
Knolls Atomic Power Lab., Schenectady, N. Y.
STRESSES IN TUBE SHEET AND TUBE SHEET RING OF THE PRIMARY HEAT EXCHANGER. J. Zickel. Aug. 6, 1952. Decl. Feb. 23, 1957. 12p. Contract W-31-109-Eng-52. $3.30(ph OTS); $2.40(mf OTS).

9834 KAPL-M-WMC-4
Knolls Atomic Power Lab., Schenectady, N. Y.
OBSERVATIONS ON AN IRRADIATED RIBBONLIKE FUEL ELEMENT. W. M. Cashin, B. Wimple, and H. L. Zoch. Oct. 31, 1956. Decl. Feb. 23, 1957. 8p. Contract W-31-109-Eng-52. $1.80(ph OTS); $1.80(mf OTS).

During in-pile cycling tests on a U–Zr alloy ribbon-type fuel element, the element became grossly deformed without concomitant cladding failure or release of fission products. Results are given of an investigation of the appearance of a cross section of the fuel in this deformed region. (B.J.H.)

9835 LRL-107
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
THE THERMODYNAMICS OF PYROCHEMICAL PROCESSING. E. L. Anderson, K. L. Mattern, W. H. McVey, and H. Schneider. Mar. 1954. Decl. Feb. 27, 1957. 14p. Contract AT(11-1)-74. $0.25(OTS).

High-temperature methods for processing power reactor fuels are discussed in the light of known thermodynamic quantities. Two illustrative flow schemes are presented which involve molten metal-molten salt transfer of plutonium. (auth)

9836 M-3743
[Hanford Engineer Works, Richland, Wash.]
ACTIVITY OF PILE DISCHARGE WATER. Interim Report on Production Test No. 105-2-P. John M. West and John J. O'Connor. Sept. 26, 1945. Decl. Mar. 7, 1957. Includes Attachments: HW-7-2618; HW-7-2619; and HW-7-2620. 14p. Contract [W-31-109-Eng-52]. (HW-3-3222). $3.30(ph OTS); $2.40(mf OTS).

Periods of excessive pile coolant activity led to an investigation of activity sources, and radioactive film deposition and discharge. The predominant activity was found to be Mn^{56}, believed to originate from an (n,p) reaction with Fe^{56}. Radioactive film was found on all surfaces in contact with the coolant; the rate of deposition being a function of time. The addition of 100 ppm citric acid to pile coolant significantly reduces film deposition. (D.E.B.)

9837 MonP-272
[Clinton Labs., Oak Ridge, Tenn.]
PHYSICS OF THE HIGH FLUX PILE. II. Mar. 27, 1947.

Decl. Feb. 11, 1957. 67p. Contract W-35-058-eng-71. $0.45(OTS).

Discussions are included on the general description, neutron flux distributions, critical data, control, U^{233} production, shielding, and heat transfer properties of the LITR. (B.J.H.)

9838 MonP-454

Clinton Labs., Oak Ridge, Tenn.

CRITICALITY STUDIES ON ENRICHED URANIUM–HEAVY WATER SYSTEMS. A. H. Snell. Dec. 15, 1947. Addendum: Aug. 1950. Decl. Mar. 7, 1957. 74p. Contract W-35-058-71. $0.45(OTS).

Addendum issued by Oak Ridge National Lab., Tenn.

The critical masses for systems consisting of U^{235} dissolved in D_2O have been studied for mean concentrations of 2.58, 5.17, and 10.35 grams of U^{235} per liter. The values obtained were respectively 1323, 930, and 869 grams U^{235} in roughly cylindrical geometry surrounded with D_2O reflector on all sides. The effects of various holes and thimbles penetrating the lattice and the reflector are given. The temperature coefficients were found to be respectively -0.85×10^{-3}, -0.59×10^{-3}, and -0.47×10^{-3} in terms of $\delta\kappa/\kappa$ per degree centigrade over the range 20-80°C. The effectiveness of a single control rod was measured in terms of distributed poison in the reactor and in terms of grams of U^{235}. The efficiency of utilization of the leakage neutrons in a ring of thorium rods in the cylindrical part of the reflector was measured. Neutron distributions are given as obtained by foil activation in the various modifications of the reactor. (auth)

9839 NAA-SR-46

North American Aviation, Inc., Los Angeles.

PHYSICS OF THE LOW POWER RESEARCH REACTOR. E. R. Cohen. Feb. 20, 1950. Decl. Feb. 27, 1957. 34p. Contract [AT-11-1-GEN-8]. $6.30(ph OTS); $3.00(mf OTS).

A homogeneous, graphite, enriched-U reactor is investigated as a function of the density of impregnation of the U. The critical mass, flux distribution, kinetics, reactivity coefficients, and effectiveness of control rods are calculated. The minimum critical mass is achieved with an impregnation density of approximately 0.010 gm cm^{-3} of U (50% U-235) and requires 2 kgms of fissionable material. Such a reactor is, however, much less safe than one with a lower impregnation density. If the U concentration is reduced to 0.005 gm cm^{-3} the volume of the core is only slightly more than doubled and the critical mass not appreciably increased, but the safety problem is considerably improved. (auth)

9840 NAA-SR-210

North American Aviation, Inc., Downey, Calif.

THE EFFECT OF FUEL PARTICLE SIZE ON THE TRANSIENT BEHAVIOR OF A HOMOGENEOUS GRAPHITE REACTOR. D. L. Hetrick. Dec. 1, 1952. Decl. Feb. 28, 1957. 20p. Contract AT-11-1-GEN-8. $0.25(OTS).

Heat transfer from the fuel to the moderator during a sudden power increase in a U-impregnated graphite-moderated reactor was investigated to determine if the use of relatively large fuel particles inhibits effective operation of the negative temperature coefficient of reactivity. Time lags in heat transfer to the moderator after the sudden introduction of 2% excess reactivity are given for the Low Power Research Reactor. The fraction of fission fragments escaping into the moderator is also given. It is concluded that radiation damage can be prevented without inhibiting the action of the temperature coefficient. (D.E.B.)

9841 NAA-SR-238

North American Aviation, Inc., Downey, Calif.

PHYSICS OF THE NAA HOMOGENEOUS GRAPHITE RESEARCH REACTOR. J. B. Ehrman and W. J. Houghton. Apr. 10, 1953. Decl. Mar. 4, 1957. 60p. Contract AT-11-1-GEN-8. $0.40(OTS).

The neutron physics involved in the engineering design of the NAA Homogeneous Graphite Research Reactor (Model S-5) are discussed. This unit has a nominal operating power of 150 kilowatts. It has a U impregnated graphite core and a graphite moderator, and it utilizes D_2O as the primary coolant. A study is made of the effect of varying the quantity z (nuclei of U^{235}/100,000 nuclei of C), in the range from 5 to 20, upon the following items: critical mass, critical volume, minimum power to yield a thermal flux of $10^{12}/cm^2$-sec at the reflector peak, and temperature coefficient of reactivity. Several improvements in design and performance result from choosing z = 15. The temperature coefficient is somewhat impaired by this choice, and the prompt neutron lifetime is decreased. For an effective core volume of 0.94×16^6 cm^3, criticality calculations are performed for four approximations of the core-reflector configuration. The most reliable result appears to be 3.53 kilograms (cold), or 4.22 kilograms, including burn-up for a core lifetime of 3.5×10^6 kilowatt hours, warm-up, fission product poisoning and experimental facilities. The total reactivity to be controlled is about $9\frac{1}{2}$ per cent, of which 2 per cent represents the allowance for experiments. Control rod dynamics and reactor runaway behavior are also discussed. It is shown that the S-5 design has a high inherent safety against runaway, which arises principally from the negative temperature coefficient of reactivity. (auth)

9842 NAA-SR-243

North American Aviation, Inc., Downey, Calif.

EPITHERMAL NEUTRON ABSORPTION IN NON-REFLECTED NATURAL URANIUM–D_2O LATTICES. M. A. Greenfield and R. H. Sehnert. May 8, 1953. Decl. Feb. 27, 1957. 23p. Contract AT-11-1-GEN-8. $0.25 (OTS).

The NAA exponential assembly has been used to study systematically natural uranium–D_2O lattices. Differences between measured and calculated values of the Laplacian for these lattices have led to consideration of epithermal neutron absorption in the analysis. Such absorption has been taken into account in a calculation of η, the number of neutrons produced per thermal absorption. This value of η is then compared with a calculation of η_{eff} in which epithermal absorptions are neglected. Representation of the slowing down phenomena is discussed. The ratio η_{eff}/η ranges from 1.016 to 1.035 for the 22 lattices considered. (auth)

9843 NAA-SR-1034

North American Aviation, Inc., Downey, Calif.

IMPROVED METHOD FOR NUMERICALLY SOLVING MULTI-GROUP REACTOR EQUATIONS. G. W. Lehman. Sept. 15, 1954. Decl. Mar. 12, 1957. 30p. Contract AT-11-1-GEN-8. $0.30(OTS).

A method for solving multi-group reactor equations which arise in the diffusion approximation is outlined. One approach is to replace the differential equation by difference equations. Complications arise in this method where more than one slowing down medium is present since the fluxes are discontinuous at the interfaces. The primary purpose of this article is to develop an exact integral expression for the neutron flux which automatically satisfies the boundary conditions. An iterative method for obtaining the fluxes and critical neutron multiplication ratio based upon the above-mentioned integral expression is given. The only approximation used in obtaining the fluxes, in addition to the use of multi-group diffusion theory as the basic model, is the use of numerical integration to evaluate the analytic expression. The equations for a two-region, two-

group spherical reactor are given in a form suitable for machine programing. The extension to more than two regions is also considered. (auth)

9844 NAA-SR-1634

Atomics International Div. North American Aviation, Inc., Canoga Park, Calif.

REACTOR SAFETY QUARTERLY PROGRESS REPORT [FOR] NOVEMBER 1955–JANUARY 1956. Stephen H. Fitch and Norman E. Huston, eds. July 15, 1956. Decl. Mar. 27, 1957. 31p. Contract AT-11-1-GEN-8. $0.35 (OTS).

The Mark II and III reactivity loss measurements have been completed. The control ratio, defined as the ratio of triggered-state to stored-state reactivity loss, is 3.0 for the Mark II design and 3.8 for the Mark III design. These values are somewhat lower than expected due to excessive reactivity loss for the stored state, caused probably by small amounts of high cross section impurities. Four test units of the production pile test (NAA-109) for studying the behavior of the safety device in the actual Hanford process tube environment have been completed. The test units, which include one dummy for practice loading, have been shipped to Hanford. A new thin slab test element, designed to reduce heat loss effects due to radiation, has been fabricated. The test element has the geometry of the cross section of a typical fuse assembly enlarged by a factor of ten. It is thicker than previous test elements to provide better heat conduction and hence is less affected by radiation loss. Data obtained from thermal transients have yielded an equivalent thermal relaxation time of 1.12 seconds, a value in good agreement with the electrical analog values of 1.04, 1.06, and 1.07 for similar geometries. An addition was made to the NAA-8 fuse assembly electrical analog representing the complete circumference of the receiving chamber and Al cladding. There was no significant change in the relaxation time, indicating that the original approximation was adequate. Thermal measurements have been made for a simulated NAA-8 assembly consisting of a full-size fuse bonded to a two-inch section of a stainless steel receiver chamber. Thermal relaxation time was 1.47 seconds, which is in fair agreement with the NAA-8 fuse assembly analog value of 1.27 seconds. The engineering tests that have been conducted are the Mark II impact test and the Mark III modified receiver pressure test. There were no external failures on the impact test, indicating a satisfactory design. The Zr receiver chambers with double thickness walls failed at 510 and 590 psi. Although these modified samples were approximately twice as strong as the nominal design chambers, they were still inadequate. Failure, as before, was along the tube seam. The design is unsatisfactory at present, but it is possible that Zr could be used if it were fabricated without a longitudinal weld. (For preceding period see NAA-SA-1568.) (auth)

9845 NAA-SR-Memo-20

North American Aviation, Inc., Downey, Calif.

CONVERSION OF IRRADIATION UNITS FOR THERMAL, NATURAL URANIUM REACTORS. R. Loftness. Oct. 13, 1950. Decl. Mar. 11, 1957. 7p. Contract AT-11-1-gen-8. $1.80(ph OTS); $1.80(mf OTS).

A chart and a table are presented for thermal natural uranium reactors, to inter-convert the irradiation units of percent burnup, Mwd/t integrated neutron flux, and the ratio of U^{235} atoms fissioned to total uranium atoms. (auth)

9846 NAA-SR-Memo-48

North American Aviation, Inc., Downey, Calif.

WATER BOILER CALCULATIONS—I. E. F. Weisner. July 25, 1951. Decl. Feb. 27, 1957. 5p. $1.80(ph OTS); $1.80(mf OTS).

Calculations of the fuel requirements for a water boiler reactor are summarized. (B.J.H.)

9847 NAA-SR-Memo-49

North American Aviation, Inc., [Downey, Calif.].

SECULAR CHANGES IN Pu PRODUCTION IN A HOMOGENEOUS REACTOR. E. R. Cohen. July 30, 1951. Decl. Feb. 27, 1957. 4p. $1.80(ph OTS); $1.80(mf OTS).

Equations are given for Pu production in a form which separates thermal from resonance capture. Plutonium production is tabulated as a function of irradiation. (B.J.H.)

9848 NAA-SR-Memo-72

North American Aviation, Inc., [Downey, Calif.].

WATER BOILER CALCULATIONS III. E. Weisner. Aug. 10, 1951. Decl. Feb. 27, 1957. 3p. $1.80(ph OTS); $1.80(mf OTS).

Calculations concerning a step increase in reactivity of the water boiler are briefly discussed. (B.J.H.)

9849 NAA-SR-Memo-92

North American Aviation, Inc., [Downey, Calif.]

SUMMARY OF PERFORMANCE CALCULATIONS FOR PuP. C. Trilling and E. Weisner. Sept. 17, 1951. Decl. Feb. 28, 1957. 6p. Contract [AT-11-1-gen-8]. Supersedes NAA-SR-Memo-67. $1.80(ph OTS); $1.80(mf OTS).

Performance calculations are presented for several systems considered as possible producers of Pu and power. (C.H.)

9850 NAA-SR-Memo-155

North American Aviation, Inc., [Downey, Calif.]

PuP PARAMETER SURVEY CALCULATIONS. E. R. Cohen. Dec. 7, 1951. Decl. Feb. 27, 1957. 7p. $1.80(ph OTS); $1.80(mf OTS).

Parameter calculations are surveyed for the PuP reactor with a 7-rod clump lattice of enriched U rods. Data are included on heat transfer characteristics, neutron flux distributions, fuel element volumes, disadvantage factors for the U rods, resonance escape probabilities, the fast effect, and other parameters. (B.J.H.)

9851 NAA-SR-Memo-188

North American Aviation, Inc., [Downey, Calif.]

HEAT TRANSFER OF PRESSURIZED WATER PuP. C. H. Robbins. Jan. 24, 1952. Decl. Feb. 27, 1957. 15p. $3.30(ph OTS); $2.40(mf OTS).

A description of the pressurized water PuP reactor is given including lattice calculations, temperatures and water flow rates. Reasons are given for various design decisions. The most economical average water temperature is the lowest which permits extraction of power. This was chosen to be 450°F. Water pressure of 2500 psia was determined by difficulties in manufacturing pressure vessels and by the condition that the water shall not boil. The maximum velocity of 30 ft/sec was chosen because a higher velocity does not appear to gain enough reactor power to balance the added problems in piping and pumping. To take advantage of the non-uniform heat generation, different velocities are used in four zones of the reactor. Flattening the central zone permits an increase in reactor power from 224 Mw to 315 Mw. A change from 20 mil Zr to 10 mil stainless steel by varying the enrichment would permit a small increase in reactor power at the expense of a small increase in water flow. (auth)

9852 NAA-SR-Memo-203

North American Aviation, Inc., Downey, Calif.

FLATTENED vs UNIFORMLY LOADED CORE. A. Benton. Jan. 30, 1952. Decl. Mar. 22, 1957. 7p. Contract [AT-11-1-GEN-8]. $1.80(ph OTS); $1.80(mf OTS).

The conversion ratio of a uniformly loaded, or unflattened reactor core, is compared with that of a flattened zone plus

matching region, or flattened core. The reactor fuel and Th elements considered are described in Memo 165. (C.H.)

9853 NAA-SR-Memo-266
North American Aviation, Inc., [Downey, Calif.]
ADDITIONAL PRELIMINARY CALCULATIONS OF FLUX DISTRIBUTIONS IN CONVERTER REACTOR CELLS. A. Benton. Mar. 24, 1952. Decl. Mar. 4, 1957. 12p. Contract [AT-11-1-gen-8]. $3.30(ph OTS); $2.40(mf OTS).

The flux distribution was calculated within a proposed unit cell of a lattice for the converter reactor. This new design consisted of Th plates brazed to the sides of the fuel elements. The results of this investigation showed that this design was unsatisfactory. Calculations and sketches of the design are presented. (C.H.)

9854 NAA-SR-Memo-323
North American Aviation, Inc., Downey, Calif.
THE EFFECT OF NON-UNIFORM LATERAL NEUTRON FLUX IN AN ASSEMBLY OF WATER-COOLED FLAT-PLATE FUEL ELEMENTS ON THE MAXIMUM HEAT FLUX AND TOTAL REACTOR POWER. H. L. Sletten. May 19, 1952. Decl. Mar. 1, 1957. 13p. Contract [AT-11-1-Gen-8]. $3.30(ph OTS); $2.40(mf OTS).

Heretofore, heat transfer calculations for the MTR type assemblies of the Converter Reactor have been based on the assumption that the neutron flux over the cross section of the assembly is uniform. We knew, of course, that a gradient of flux exists in the assembly due to a net diffusion of neutrons outward to produce U^{233} from Th^{232} in adjacent feed assemblies, and NAA-SR-Memos-174 and 266 have indicated that the ratio of minimum to maximum thermal flux in an assembly of this type may be as low as 0.75. The effects of this cellular variation of flux upon the total reactor power is evaluated. Specifically, for CR-1 (Converter Reactor design designation in NAA-SR-Memo-200) with a lateral flux distribution as described in the above memos, the total reactor power is 6% less than the power calculated on the basis of an assumed uniform assembly flux. (auth)

9855 NAA-SR-Memo-351
North American Aviation, Inc., Downey, Calif.
DISTRIBUTION OF HEAT PRODUCTION FOR THE CONVERTER REACTOR. W. C. Cooley. June 6, 1952. Decl. Feb. 27, 1957. 4p. Contract [AT-11-1-GEN-8]. $1.80(ph OTS); $1.80(mf OTS).

9856 NAA-SR-Memo-621
North American Aviation, Inc., Downey, Calif.
SPECIFIC ACTIVITY AND SHIELDING REQUIREMENTS OF THE PRIMARY COOLANT CIRCUIT FOR THE UPR. Ralph Balent. Mar. 17, 1953. Decl. Feb. 28, 1957. 19p. Contract [AT-11-1-GEN-8]. $3.30(ph OTS); $2.40(mf OTS).

The various sources of activity in the primary coolant of the UPR are examined, and the required shielding for 36 in. diameter and 72 in. diameter pipes is examined. The N^{16} decay governs the shielding requirements for approximately the first minute after leaving the reactor core. There is essentially no difference in the shielding requirements for either the 36 in. or 72 in. pipe. Immediately upon leaving the reactor, a 5 ft concrete shield is needed for either pipe. After 1-5 min the N^{16} has decayed to a very low level and the shielding requirement is a constant for the rest of the coolant loop and is governed by the equilibrium concentration of the various dissolved impurities. The summation of the activities from all the impurities require a 1 ft concrete shield. If earth is to be used instead of concrete for the shielding, the required thickness is approximately twice the value given here for concrete. The possibility of using either Admiralty Metal or Stainless Steel for the tubes in the heat exchangers was investigated. It was assumed that the corrosion rate would be the same for these two materials. Under this assumption it is shown that there would not be any appreciable difference in the water activity from the use of either material for the tubes of the heat exchanger. (auth)

9857 NAA-SR-Memo-640
North American Aviation, Inc., [Downey, Calif.]
UPR INSTRUMENTATION. F. L. Fillmore and E. Matlin. Mar. 27, 1953. Decl. Mar. 1, 1957. 24p. $6.30(ph OTS); $3.00(mf OTS).

The essential features of the control and instrumentation for the UPR are discussed. A block diagram of the proposed system is included. The operating procedure for reactor start-up is described. Calculations on the hazard involved in a start-up accident are given. (auth)

9858 NEA-5401
Diamond Alkali Co., Cleveland; Foster Wheeler Corp., New York; and Pioneer Service and Engineering Co., Chicago.
FEASIBILITY REPORT ON A FLUIDIZED SOLIDS REACTOR SYSTEM. Jan. 1954. Decl. Feb. 28, 1957. 37p. $0.35(OTS).

The feasibility of power reactors using a circulating gas-fluidized solids system is discussed. Carbon moderator particles, impregnated with U^{233}, flow as a settled bed through a carbon reactor core. A fluidized ThC_2 blanket is maintained outside of the core. The system was found to be impractical for the following reasons: (1) a boiling fluidized bed undergoes rapid and large bulk density changes, (2) a quiescent bed cannot be maintained in a reactor core, (3) a uniform particle density is difficult to maintain at high temperature, (4) the attrition rate of graphite is excessive, and (5) the temperature coefficient of reactivity is insufficient to provide adequate control. (K.S.)

9859 NYO-3924
Kidde (Walter) Nuclear Labs., Inc., Garden City, N. Y.
LIGHT WATER MODERATED REACTORS IN CROSS FLOW. Sept. 30, 1953. Decl. Feb. 27, 1957. 90p. Contract AT(30-1)-1374, Task 1. (WKNL-15). $0.50(OTS).

Results are reported from exponential measurements at BNL on H_2O-reflected lattices of U rods 0.750 in. in diam., 4 ft long, and jacketed with 30 mils of Al. Experiments are described for determination of the average heat-transfer coefficient, the local heat-transfer coefficients, and the onset of nucleate boiling for H_2O flowing across a bank of staggered rods at high Reynolds numbers; the effect of thermal cycling on rod-to-jacket transfer for an unbonded jacket was also investigated. Results are presented from an investigation of the hydrodynamic effect of high-velocity H_2O flowing perpendicularly to a bank of rods arranged in a triangular lattice. The preparation of long Zr-clad U fuel rods and the corrosion testing of samples of annealed Zr tubing in distilled water and solutions of H_3BO_3 and $K_2B_4O_7$ are discussed. (L.M.T.)

9860 ORNL-402
Oak Ridge National Lab., Tenn.
THE ORNL SHIELD TESTING FACILITY. C. E. Clifford. Nov. 4, 1949. Decl. Mar. 7, 1957. 11p. Contract W-7405-eng-26. $1.80(ph OTS); $1.80(mf OTS).

A general description is given of the Lid Tank Facility. (B.J.H.)

9861 ORNL-730
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR EXPERIMENT FEASIBILITY REPORT. July 6, 1950. Decl. Mar. 2, 1957. 108p. Contract W-7405-eng-26. $0.55(OTS).

9862 ORNL-951
Oak Ridge National Lab., Tenn.
PRODUCTION OF FUEL ASSEMBLIES FOR THE MATERIALS TESTING REACTOR MOCK-UP CRITICAL EXPERIMENTS. Period Covered: November 1, 1949 – March 1, 1950. C. D. Smith, F. W. Droston, and F. Kerze, Jr. Apr. 9, 1951. Decl. Mar. 2, 1957. 63p. Contract W-7405-eng-26. $0.40(OTS).

The techniques involved in the fabrication of MTR fuel assemblies and fuel sections of shim rods are summarized. Specifications, materials, equipment, layout and organization, assembly of components, inspection, final fabrication, U accountability, and health physics operations are discussed and illustrated. (D.E.B.)

9863 ORNL-1057
Oak Ridge National Lab., Tenn.
HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MAY 15, 1951. W. E. Thompson, Comp. Oct. 10, 1951. Decl. Mar. 12, 1957. 147p. Contract W-7405-eng-26. $22.80(ph OTS); $7.20(mf OTS).

Design and construction progress on the HRE catalytic recombiner, reactor tank assembly, reflector-compartment piping, emergency cooling system, shield assembly, bellows valve operator, and leak detection system is illustrated. Static corrosion studies reported include passivation of stainless steels (SS) used to contain UO_2SO_4 at 250°C by pretreatment; effect on the UO_2SO_4 solution of long-term exposure at 250°C to treated and untreated SS; corrosion of Ti and Zr–5% Sn in 250°C UO_2SO_4; corrosion of Zr, Ti, and 347 and 309 Nb SS, in 150°C UO_2SO_4; corrosion of Ti, types 309, 304, 321, 347, and 316 SS, Inconel, Durimet 20, Carpenter 20, Worthite, and Illium R in 150° UO_2F_2; and corrosion of SAE 1030 carbon steel in distilled water, H_2O_2, H_2O_2 plus trisodium phosphate inhibitor, and when galvanically coupled to 347 SS. The effect of numerous variables on these corrosion studies is discussed. Electrochemical corrosion in a cell consisting of passivated SS electrodes in UO_2SO_4 at 100°C has been studied. The status of stability and corrosion experiments under pile irradiation is summarized. Studies on the recombination of H_2 and O_2; removal of I_2 from the gas stream, electromagnetic densitometer calibration, vortex stability in the letdown heat exchanger, recirculation pump bearings, and mixing characteristics in the HRE core are reported. Additional data on the $UO_2(NO_3)_2$–H_2O, UO_3–H_3PO_4–H_2O, and $AgSO_4$–H_2O phase systems are tabulated. Further investigations on UO_3 and ThO_2 slurries, continuous fuel processing, and slurry and boiling reactors are summarized. (G.Y.)

9864 ORNL-1075
Oak Ridge National Lab., Tenn.
AN EXPERIMENTAL DETERMINATION OF FISSION PRODUCT HEATING AFTER SHUTDOWN OF THE LOW INTENSITY TRAINING REACTOR. S. E. Beall. Sept. 25, 1951. Decl. Mar. 4, 1957. 23p. Contract W-7405-eng-26. $0.25(OTS).

Experiments have been performed in the Low Intensity Training Reactor to measure the rate of fission product heat dissipation from the fuel pieces after loss of water. The measurements provide evidence that: the fission product power in a single fuel piece containing 140 gm U-235 can be expressed as $q = 14\ P\ [t^{-0.2} - (t + T)^{-0.2}]$ Btu/hr where P = reactor power before shutdown in kilowatts, t = time after shutdown in seconds, and T = operating time before shutdown in seconds; that as much as 75% of the fission product heat is lost by conduction to other parts of the reactor before the fuel reaches a maximum temperature; and that the LITR may be operated at a power level of about 1500 to 2000 kw without danger of melting the fuel even if the cooling water is suddenly removed. (auth)

9865 ORNL-1726
Oak Ridge National Lab., Tenn.
CRITICAL MASS STUDIES. PART VII. AQUEOUS URANIUM SLURRIES. D. F. Cronin and Dixon Callihan. July 1, 1954. Decl. Mar. 2, 1957. 33p. Contract W-7405-eng-26. $0.30(OTS).

A series of experiments was performed with aqueous slurries of UO_3, enriched to 93% in U^{235}, as a preliminary study of the nuclear stability of critical and near critical suspensions. The slurries were contained in a 12" diameter cylinder and were mixed by a centrally located variable speed stirrer in a manner giving small surface disturbances. Critical masses were measured at concentrations ranging from 40 to 300 g U/L at several stirrer speeds. The dependence of the critical parameters of well mixed slurries on concentration is essentially the same as found earlier with solutions. The critical masses of imcompletely mixed slurries were measured to be 5% to 10% less than those of well mixed ones under otherwise similar conditions, an amount not strongly dependent on the chemical concentration. Neutron activity excursions of corresponding magnitude were observed in near critical quantities of the slurry following stopping the stirrer in a well mixed system and, in some instances, after starting agitation of one which had settled. Although the activity changes lagged the alterations of the stirrer speed, in most cases requiring a few tens of seconds to develop, the magnitudes of the excursions are sufficient to warrent consideration of this effect in the safety features of reactors designed to use slurry fuels. The critical mass of a $(UO_2)F_2$ solution in the same vessel increased monotonically as the stirrer speed was increased over the same range, the total change being about 3%, due in part at least, to some vortex formation at the higher speeds. It is concluded that the activity excursions in the slurry result from changes in the U distribution more complex than those which occur in settling alone. An analysis of the distribution, relating it to the neutron activity, has not been made so the results of the experiments are not completely interpretable. (auth)

9866 ORNL-1761
Oak Ridge National Lab., Tenn.
PRELIMINARY COST ESTIMATION: CHEMICAL PROCESSING AND FUEL COSTS FOR A THERMAL BREEDER REACTOR POWER STATION. E. D. Arnold, A. T. Gresky, R. J. Klotzbach, and A. R. Irvine. Feb. 23, 1955. Decl. Feb. 25, 1957. 64p. Contract W-7405-eng-26. $0.50 (OTS).

Results of a cost estimate for the chemical process associated with a three-reactor power station of a proposed homogeneous breeder system, the K-23 reactor, indicate that the chemical processing cost will be 1.06 mills per kilowatt-hour of electricity. This cost is based on current Thorex technology for a limited size plant, and is not representative of the optimum plant size which would be integrated in a larger power station. The unit fuel cost, which includes reactor inventories, fuel consumption, and losses in addition to the chemical processing cost, is estimated at 2.19 mills per kilowatt-hour of electricity, or $12.26 per megawatt-day of heat. (auth)

9867 ORNL-1794(Vol. I)
Oak Ridge National Lab., Tenn.
THE OAK RIDGE NATIONAL LABORATORY RESEARCH REACTOR SAFEGUARD REPORT. VOLUME I. F. T. Binford, ed. Oct. 7, 1954. Decl. Mar. 2, 1957. 103p. Contract W-7405-eng-26. $0.55(OTS).

The proposed ORNL Research Reactor is designed to serve as a general purpose research tool delivering a maximum thermal flux of 8×10^{13} n/cm^2-sec at the initial power level of 5 Mw. Operation at power levels up to 10 Mw is proposed for such times as sufficient cooling capacity is available to handle the increased heat load. The reactor will use MTR type fuel elements and Be reflector pieces in a 7×9 grid with moderation and cooling provided by forced circulation of demineralized water. The reactor tanks are submerged in a barytes concrete pool, filled with water, which serves as a biological shield. Experimental facilities include two 18" diameter Engineering Test Facilities and six 6" diameter beam holes. In addition, access to the core is available through the water of the pool. (auth)

9868 ORNL-1836

Oak Ridge National Lab., Tenn.

A MONTE CARLO ESTIMATION OF THE HIGH ENERGY NEUTRON FLUX DISTRIBUTION IN THE ORNL GRAPHITE REACTOR. Jack Moshman. Feb. 23, 1955. Decl. Mar. 1, 1957. 32p. Contract W-7405-eng-26. $0.35(OTS).

Three representative regions within the ORNL graphite reactor lattice were chosen for investigation of the fast neutron flux. CPC and UNIVAC calculations were carried out for 14 energy intervals between 0.025 and 10 Mev. (K.S.)

9869 ORNL-1895

Oak Ridge National Lab., Tenn.

HOMOGENEOUS REACTOR PROJECT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING APRIL 30, 1955. H. F. McDuffie and D. C. Kelly, comps. July 14, 1955. Decl. Mar. 2, 1957. 220p. Contract W-7405-eng-26. $1.10(OTS).

Homogeneous Reactor Test. Final flow diagrams for the HRT system are presented. The HRT cell is being modified so that it can contain a pressure of 30 psig which might result from simultaneous rupture of fuel and blanket steam systems. Designs of a blast shield and fuel low pressure assembly have been completed, and the items are under construction. The effect of prompt neutron lifetime upon reactor safety was investigated for the HRT. The allowable rate of decrease in steam pressure on the shell side of the heat exchanger was calculated. The effects of overpressure and temperature upon the stability of the HRT against "walk-away" are reported. Design and testing of a number of pumps are reported. The first phase of the HRT dump test, which was carried out to obtain fundamental information on the pressure-time curves of the pressure vessel, pressure drops through the discharge line and dump valve, and mass flow rates, is nearing completion. The HRT mock-up loop completed the water phase of its operation with excellent performance. A pneumatic weigh system for high accuracy inventory of the dump and condensate tanks was selected and procured. A cylinder-type operator featuring nylon O-ring seals has been selected as the HRT valve actuator. Thorium Breeder Reactor. An economic study of one-region Th breeder reactors was completed, and results are reported. The process flow diagram of the reactor plant has been completed and is presented. A possible reactor and power plant layout is described. The heat balance for the turbine cycle was completed. Preliminary designs of several reactor components are presented, including: reactor vessel and core pressurizer; core and blanket circulating pumps, steam generators, and dump tanks; recombiners; and the fission-gas disposal system. Corrosion. The results of pump-loop tests on the corrosive effects of UO_2SO_4 on stainless steels, titanium alloys, Zircaloy, platinum, Inconel, and Stellite are presented in tabular form. It was demonstrated that the introduction of large amounts of a salt of sulfuric acid into a UO_2SO_4 solution reduced the corrosion of stainless steel substantially at all flow rates. A survey of the corrosion of stainless steel, platinum, and Ti and Zr alloys by various mineral acids at high temperatures is reported. Determination of the sedimentation rates of ThO_2 slurries at elevated temperatures is continuing. The design and operation of in-pile loops circulating UO_2SO_4 or a ThO_2 slurry are described. Results of in-pile loop corrosion tests on Zircaloy and stainless steel samples and in-pile bomb corrosion tests on Ti and Zircaloy samples are reported. Data are also reported on the corrosion of precipitation-hardening stainless steel by simulated reactor environments, the corrosion of stainless steel, Ti, and Zircaloy by ThO_2 slurries, and the effect of CO_2 in ThO_2 slurries. Engineering Development. Several types of heterogeneous catalysts for H_2-O_2 recombination were evaluated in the high-pressure recombiner loop. The performances of a 14-in. gas separator with singly curved vanes and several small pumps are reported. Operation of a 100-gpm loop with ThO_2 slurry at 290°C indicated no greater over-all attack rate on stainless steel than in a comparable test at 250°C. Corrosive effects, thickening, and the effects of additives are being studied in ThO_2 slurry loop tests. Various commercial valves are being tested for use with slurries, and results are summarized. Slurry tests in the blanket vessel mockup were completed. Pressure drops across the blanket vessel and homogeneity of the circulating stream were determined for various flow rates. Quantitative measurements of the accumulation of solids in a vertical pressurizer have been made as a function of steam withdrawal rate. Construction of the HRT blanket mockup has started, and a flowsheet is presented. A method for correlating both volume-boiling and surface-boiling vapor slip velocities was developed and compared with experimental data. Results of stress-corrosion cracking tests of austenitic stainless steels in boiling magnesium chloride solution are presented. Metallurgical investigations to determine the cause of embrittlement of commercial Ti were continued. Zircaloy-2 multiple-break impact specimens from an in-pile loop were tested, and data are presented in graphical form. The results of tensile tests on irradiated austenitic stainless steels are reported. Chemical Engineering Development. Experiments have been continued on the precipitation of neodymium sulfate on hot metals and the solubility of neodymium sulfate in HRT fuel solutions, and results are summarized. A procedure for purifying Nd^{147} is described. Iodine can be removed from fuel solution under reactor conditions by placing metallic Ag in the vapor phase. It was found that connecting the underflow pot of a hydroclone directly to the hydroclone underflow port eliminated the necessity for recycling the hydroclone underflow. Hydroclone performance using this arrangement, called the "induced underflow pot system", is described. Corrosion studies indicate that a material more resistant then stainless steel may be required for the hydroclone. The HRT chemical processing plant and instrumentation for the plant are described. The study of plutonium chemistry under simulated reactor conditions has included Pu adsorption on metals, the rate of PuO_2 precipitation, and separation of PuO_2 with a hydroclone. Removal of Pa and U from the ThO_2 blanket by leaching is being investigated. Zircaloy-2 corrosion rates in ThO_2-ThF_4 slurries are reported. Supporting Chemical Research. Phase studies of aqueous uranyl sulfate and UO_3 systems are reported. Exploratory work on the adsorptive properties of various inorganic materials for ions in aqueous solutions was continued. Principal emphasis centered on the properties of zirconium hydroxide, zirconium phosphate, zirconium tungstate, and thorium hydroxide. Research in analytical chemistry has been directed toward determination of trace elements which are present as products of

corrosion or as additives in solutions of uranyl sulfate and slurries of thorium oxide. (B.J.H.)

9870 TID-10020

Argonne National Lab., Lemont, Ill.

EXPERIMENTS ON ZERO POWER REACTOR (ZPR-I). PART III. F. H. Martens. Mar. 1954. Decl. Mar. 12, 1957. 286p. Contract W-31-109-eng-38. $1.10(OTS).

Three phases of current experimental research on the rebuilt ZPR-1 are reported in detail. The first part consists of a study of several control rod designs and their effect on reactor criticality. The minimum critical mass was determined for all cases. Flux maps and plots are given showing the effect of control rod design on thermal flux behavior. Phase two describes experiments with the full size core. Multiplication measurements were performed in order to determine k_{eff} in the shutdown condition. Flux contour maps of the full size core were obtained by Mn-activation techniques for 2 levels in the reactor. The final phase presents data on flux distribution when B is added to the moderator water. Neutron and γ fluxes were measured at various points outside the reactor, in connection with proposed shielding design. It was also found that high-radiation fields had an adverse structural effect on the UO_2–polystyrene fuel mixtures used in the ZPR-1 critical assembly. (B.J.H.)

9871 TID-10055

Knolls Atomic Power Lab., Schenectady, N. Y.

A PROCEDURE FOR XENON CALCULATIONS. P. L. Hofmann, H. Hurwitz, Jr., and E. Wachspress. Sept. 30, 1953. Decl. Mar. 16, 1957. 14p. Contract W-31-109-Eng-52. $0.25(OTS).

A procedure is given for the determination of Xe concentration and its effect on reactivity during reactor operation and after shutdown. The techniques outlined were devised to be particularly applicable to intermediate spectrum reactors. In general, the approach has relied—as far as terminology and formalism are concerned—on multigroup calculation methods. Particular emphasis has been placed on steady state operation and peak Xe override. For both of these conditions methods are described for calculating the Xe concentration and its effect on reactivity. (auth)

9872 TID-10064

Knolls Atomic Power Lab., Schenectady, N. Y.

FAST FISSION EFFECT IN URANIUM–WATER LATTICES. J. W. Weil. Jan. 15, 1954. Decl. Mar. 7, 1957. 22p. Contract W-31-109-Eng-52. $0.30(OTS).

Methods of calculating the fast effect, ϵ, are described for heterogeneous reactors with low ratios of moderator to fuel. A discussion of present information of inelastic scattering of fission neutrons in U is included. Results of the calculations, while in general agreement with experiments on H_2O-U lattices at BNL, nevertheless show a significant deviation from the measured values. This discrepancy is discussed in some detail. (auth)

9873 TID-10076

California Research and Development Co. Livermore Research Lab., Livermore, Calif.

REACTOR DESIGN EVALUATION PROGRAM. Quarterly Progress Report 2—April Through June 1953. June 1954. Decl. Mar. 4, 1957. 16p. Contract AT(11-1)-74. $0.25 (OTS).

To carry out the Reactor Design Evaluation Program, the work has been divided into the following parts: collection of basic technical and economical data; evaluation of reactors for power-only production; and, evaluation of processes for FM production. The major accomplishment of this quarter was the completion and issuance of a report entitled "Initial Report on Nuclear Power Plant Economics." The balance of the work was done on further development of economic data; initial studies of optimum steam temperatures for power plants; continuation of power-only reactor evaluations; further work on power reactor logistics; and preparation of a preliminary screening report of ten dual purpose reactor proposals to determine their utility under private ownership. (auth)

9874 UCRL-1663

California. Univ., Berkeley. Radiation Lab.

FLUX IN A RECTANGULAR CAVITY. M. Rosenblum. Feb. 4, 1952. Decl. Mar. 4, 1957. 5p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80(mf OTS).

Calculations are developed for determining whether local high- or low-neutron densities occur at the corners of a cubical cavity in a graphite lattice. No abnormal neutron densities are indicated. (D.E.B.)

9875 WAPD-CP-156

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PROPOSAL FOR DYNAMIC IN-PILE CORROSION TESTS AT MTR. E. S. Lembersky and Warren Witzig. Dec. 8, 1953. Decl. Mar. 5, 1957. 23p. Contract AT-11-1-GEN-14. $4.80(ph OTS); $2.70(mf OTS).

Loop design for simultaneous study of corrosion and radiation effects on fuel elements is proposed. The value of such experiments is discussed, and the expected contamination problems analyzed. (D.E.B.)

9876 WAPD-FE-867

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

PENTREX EXAMINATION OF SMALL DIAMETER ZIRCALOY-2 CLAD COMPONENTS. A. E. Oaks. June 10, 1955. Decl. Mar. 5, 1957. 24p. $0.35(OTS).

Studies were made on the effect of variations in penetration time, penetrant temperature, emulsification temperature, and wash water temperature on the sensitivity and reproducibility of pentrex, and fluorescent penetrant. From these, standard procedures, including pre- and post-cleaning methods, having been developed for the inspection of Zircaloy-2 tubes used for PWR fuel elements. (auth)

9877 WAPD-FE-919

Westinghouse Electric Corp. Atomic Power Div., Pittsburgh.

FABRICATION OF PWR BUNDLE IN-PILE TESTS. T. J. Kisiel. Aug. 8, 1955. Decl. Mar. 5, 1957. 20p. $3.30(ph OTS); $2.40(mf OTS).

Two types of PWR in-pile bundles were made. The MTR HT-1 bundles are designed to study the possibility of progressive failure in rod bundles. Four bundles with UO_2 fuel rods were made, each with its distinctive type of defect rod; a bent rod contacting two adjacent rods, a rod with holes drilled through the cladding, a water-logged rod, and a bent, water logged rod whose cladding was weakened by means of slots at the contact points. All the HT-1 bundles are tube sheet designs with a hook for insertion and removal from the loop. The CR-5 test is a 6 months proof test in the Chalk River Central Thimble Loop of two PWR-type bundles in a Zircaloy-2 flow director. The bundles are made of two designs, the welded tube sheet and the mechanical egg crate types. (auth)

9878 WAPD-FE-979

[Westinghouse Electric Corp. Atomic Power Div., Pittsburgh].

SELECTION OF PWR FUEL ROD BUNDLE DESIGN. W. L. Frankhouser. [Aug. 1955]. Decl. Mar. 5, 1957. 4p. $3.40(ph OTS); $2.40(mf OTS).

With this is bound, with separate paging: PROPOSED TESTS FOR EVALUATION OF PWR FUEL ROD BUNDLES AND NATURAL FUEL ASSEMBLIES. W. L. Frankhouser. [Aug. 1955]. 6p. (WAPD-FE-950).

Welded fuel rod bundles for the PWR were selected be-

cause they are not susceptible to fretting corrosion, they offer more structural rigidity, machining to dimensional tolerance is easier, and shrinkage is uniform. Tests are proposed for PWR fuel rod bundles and natural fuel assemblies. (F.S.)

PARTICLE ACCELERATORS

9879 CF-56-9-6
Oak Ridge National Lab., Tenn.
POSSIBLE ACCELERATOR CHOICES FOR ORNL. T. A. Welton. Sept. 1, 1956. 10p. Contract [W-7405-Eng-26]. $1.80(ph OTS); $1.80(mf OTS).

Possible choices of accelerators for ORNL are reviewed. The advantages and disadvantages of the various proposals are listed. (B.J.H.)

9880 LRL-53
California Research and Development Co. Livermore Research Lab., Livermore, Calif.
MTA PROJECT ACCELERATOR QUARTERLY PROGRESS REPORT [FOR] SEPTEMBER THROUGH NOVEMBER 1952. Dec. 1953. Decl. Mar. 2, 1957. 35p. Contract AT(11-1)-74. $6.30(ph OTS): $3.00(mf OTS).

Preliminary testing and operation during the first five runs of the MTA are reviewed. Results from these tests have indicated a need for further component modifications in order to realize full design characteristics. Drift-tube sparking presently constitutes the limiting factor in full-power operation. Programs have been initiated to study improvements in the ion source, oscillator, vacuum system, instrumentation, and drift tubes. (K.S.)

9881 LWS-24317(Del.)
California Research and Development Co., Livermore, Calif.
THE EFFECTS OF REFLECTOR MATERIAL AND LATTICE THICKNESS ON URANIUM–WATER LATTICES. W. H. Harker. Mar. 18, 1952. Decl. with deletions Feb. 27, 1957. 38p. (CRD-T4A-93(Del.)). $6.30(ph OTS); $3.00(mf OTS).

A series of two energy group, two medium calculations was made to determine the best choice of reflector materials for use with H_2O-moderated lattices. The lattice used for the calculations is light-water moderated and fueled with 1-in. diam. U rods on a $1\frac{1}{4}$-in. triangular spacing. Lattice thicknesses of 20, 25, 35, and 40 cm, backed by reflectors of D_2O, H_2O, Be, BeO, Fe, graphite, and a H_2O–Fe mixture were studied. Complete results are given in graphical and tabular form. (B.J.H.)

9882 LWS-24610
California Research and Development Co., Livermore, Calif.
DESIGN CONSIDERATIONS FOR THE WATER-COOLED CLAD URANIUM PLATE, PRIMARY AND SECONDARY TARGETS OF THE A-12. J. E. Mahlmeister and Karl Bernstein. Oct. 13, 1952. Decl. Feb. 26, 1957. 21p. (CRD-T2B-83). $4.80(ph OTS); $2.70(mf OTS).

Methods are described which were used for the prediction of heat transfer and flow conditions in water-cooled uranium primary and secondary Materials Testing Accelerator targets. Data are presented graphically. The specific data presented are considered out of date but the basis for the calculations has been used for later determinations. (C.H.)

9883 UCRL-1121
California. Univ., Berkeley. Radiation Lab.
CALCULATION OF K_∞ IN A URANIUM–GRAPHITE LATTICE. F. Adelman. Feb. 8, 1951. Decl. Mar. 6, 1957. 2p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80(mf OTS).

Equations are developed for the calculation of k_∞ for MTA lattices having various degrees of depletion. (D.E.B.)

9884 UCRL-1125
California. Univ., Berkeley. Radiation Lab.
IV. PRODUCTION AND HEATING. Harold Brown. Feb. 12, 1951. Decl. Mar. 5, 1957. 3p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80(mf OTS).

Calculations are developed which indicate the Pu production in an infinite lattice. (D.E.B.)

9885 UCRL-1847
California. Univ., Berkeley. Radiation Lab.
NEUTRON STREAMING IN A CYLINDRICAL CAVITY. R. LeLevier. June 4, 1952. Decl. Mar. 5, 1957. 5p. Contract [W-7405-eng-48]. $1.80(ph OTS); $1.80(mf OTS).

9886 UCRL-2194
California. Univ., Berkeley. Radiation Lab.
MTA QUARTERLY PROGRESS REPORT [FOR] DECEMBER 1952, JANUARY, FEBRUARY 1953. Apr. 27, 1953. Decl. Mar. 6, 1957. 25p. Contract W-7405-eng-48. $4.80(ph OTS); $2.70(mf OTS).

Progress on the MTA project is reported as follows: development of a high-performance power tube capable of efficient operation up to 50 Mc; application of strong-focusing techniques to the design of a 48-Mc pre-accelerator; experimental results on the spiral-beam bunching of accelerator beams; general studies of strong focusing; determination of the cross section for producing 1 neutron from U bombarded by 190-Mev deuterons; determination of neutron yields from U targets of varying thickness bombarded by 190-Mev deuterons; measurement of the energy spectrum of deuterons stripped from He^3 and the resultant neutron yield; and neutron yield determinations from deuteron bombardment of 2 ft × 2 ft targets of U, Be, and Al-Be. (J.E.D.)

9887 UCRL-2474
California. Univ., Berkeley. Radiation Lab.
MTA PROGRESS REPORT [FOR] JUNE THROUGH NOVEMBER, 1953. Feb. 8, 1954. Decl. Mar. 25, 1957. 46p. Contract W-7405-eng-48. $7.80(ph OTS); $3.30 (mf OTS).

Progress in the design of the 49-Mc accelerator is reported. Accelerator design was directed principally to magnetic focusing techniques, beam dynamics, and oscillator tube performance testing. A critical study of alternating-gradient focusing has shown that a very sensitive dependence on local field adjustment is present in the system, which is sufficiently critical to warrant continuation of solenoidal focusing. RCA A2332 tubes have been tested at 49 Mc c-w, indicating that successful continuous operation is feasible at 20 kv, delivering 600 kw. Improved degassing methods and several minor modifications show promise of higher power operation. Target research activities have been directed primarily to the determination of neutron yield as a function of deuteron and proton energy, and the measurement of nuclear cross sections involved in the process of neutron production. The method previously reported for producing 320-Mev deuterons from stripped 500-Mev He^3 particles has been refined. The energy distribution of the 320-Mev stripped deuteron beam has been studied. Neutron, proton, and deuteron inelastic scattering cross sections are reported for several metals. An empirical relationship has been developed from these data for the dependence of inelastic scattering cross sections upon the type of bombarding particle and target nucleus. (J.E.D.)

9888 Y-584
Oak Ridge National Lab., Y-12 Area, Tenn.
ELECTROMAGNETIC RESEARCH DIVISION PROGRESS REPORT [FOR] JANUARY 1, 1950 TO MARCH 31, 1950. R. S. Livingston. Mar. 31, 1950. Decl. Mar. 5, 1957. 82p. Contract W-7405-eng-26. $13.80(ph OTS); $4.80 (mf OTS).

An increase in ion output, exceeding fivefold, resulting from the substitution of grid structures for the conventional ion acceleration system of the calutron has indicated the value of expanding research on this new principle. Investigation of proposed means of separating these high currents has been started. The magnet for the 86-in. cyclotron has been energized, just six months from ground-breaking. Most of the electrical supplies and control equipment has been built and tested on an electrical model of the dee-system. A material balance on the U^{234} separation process shows that the 1.5 grams of 94% assay material was obtained with a dispersal of no more than 0.4% per cycle. First stage separation of irradiated uranium containing about 1% U^{236} has yielded product above 30% concentration. Procedures for the next purification stage have been worked out. A neutron flux of $1.2 \times 10^9/cm^2/sec$ has been reached with the D-D reaction spectrograph source. A preliminary mass measurement, using time-of-flight principle, gave the mass difference between Cl^{35} and Cl^{37} with an accuracy of ±0.05 mass units. (auth)

RADIATION EFFECTS

9889 ANL-5003
Argonne National Lab., Lemont, Ill.
THE USE OF THE ELECTRICAL CONDUCTIVITY OF GRAPHITE AS A RADIATION DAMAGE AND FLUX MONITOR. IMPLICATIONS OF THE RESULTS TO THE GENERAL THEORY OF RADIATION DAMAGE. W. L. Primak and L. H. Fuchs. July 13, 1953. Decl. Feb. 11, 1957. 58p. Contract W-31-109-eng-38. $0.40(OTS).

A method for monitoring radiation damage irradiations using the change of the electrical conductivity of graphite is described. Results of monitoring operations in a number of locations are given. An attempt is made to explain the damage rate found in the converter at CP-3'. An estimate of the flux spectrum in VT-4 and the converter of CP-3' is made and the damage rate is computed according to the theories of Seitz and James. It is found that the experimental results are more in accord with the predictions of Seitz. A number of corrections to James' report, ORNL-307, are noted in an appendix. It is shown that some information about the fast flux spectrum can be inferred from the radiation damage rate. (auth)

9890 ORNL-115
Oak Ridge National Lab., Tenn.
ON THE EFFECTS OF NUCLEAR RADIATIONS ON SOLIDS. Sidney Siegel. July 1, 1948. Decl. Mar. 8, 1957. 47p. Contract W-7405-eng-26. $7.80(ph OTS); $3.30(mf OTS).

The types of radiation present in a nuclear reactor are summarized, and the mechanisms by which energy is transferred from the radiations to the reactor materials are discussed. The changes observed in irradiated metals, semiconductors, graphite, crystals, and fissionable materials are described. The rate of change of properties in materials containing fissionable nuclei and subject to fission fragment bombardment is markedly higher than that due to neutron recoils. Some experiments suggest that metals are less damaged than similar insulating materials. (M.P.G.)

9891 ORNL-1938
Oak Ridge National Lab., Tenn.
THE EFFECTS OF REACTOR IRRADIATION ON THORIUM–URANIUM ALLOY FUEL PLATES. R. M. Carroll. Sept. 7, 1955. Decl. Feb. 28, 1957. 33p. Contract W-7405-eng-26. $0.35(OTS).

Several plates of 98.7% Th–1.2% U^{235} (clad in aluminum) were irradiated in the MTR for an integrated flux of 2.6×10^{21} neutrons/cm^2. Although these samples represent an early development in bonding of aluminum to thorium and there are better methods at present, the bond proved to be quite strong and both clad and core were dimensionally stable under irradiation. The production of U^{233} was as much as theory would indicate, and the total amount of fissionable material after irradiation and after decay of the Pa^{233} was greater than before irradiation. A fuel element of this nature appears to offer excellent potentialities from the standpoint of radiation stability. (auth)

SHIELDING

9892 BMI-71
Battelle Memorial Inst., Columbus, Ohio.
ENGINEERING PROPERTIES OF HIGH-DENSITY CONCRETES. M. J. Snyder, C. A. Burkart, and J. W. Clegg. May 1951. Decl. Feb. 13, 1957. 59p. Contract W-7405-eng-92. $0.40(OTS).

Methods were developed for measuring various engineering properties of high-density concretes. Data are presented on selected thermal properties of a limonite-iron concrete and the mechanical properties of the limonite-iron concrete, a magnesium oxychloride concrete, and a Portland cement concrete containing steel punchings and shot (no limonite). The properties which were determined included: thermal expansion, thermal conductivity, resistance to thermal stress, heat liberation during setting, effect of temperature and humidity on equilibrium water content, gas permeability, dimensional stability, compressive strength, modulus of elasticity, flexural strength, bonding strength, and shear strength. (auth)

9893 KAPL-76
Knolls Atomic Power Lab., Schenectady, N. Y.
THE ECONOMICAL DISTRIBUTION OF GAMMA RAY ABSORBING MATERIAL IN A SPHERICAL PILE SHIELD. Lewi Tonks and Henry Hurwitz, Jr. June 8, 1948. Decl. Feb. 27, 1957. 23p. Contract W-31-109-eng-52. $0.25 (OTS).

A simple shielding problem is set up in which a spherical shield made up of a substance which slows down and absorbs neutrons is loaded with a space-variable amount of heavy atoms to absorb photons. The photons originate both in the pile and from the absorption of neutrons in the shield. The heavy atoms absorb no neutrons and do not give rise to any gamma rays. The distribution of heavy atoms is calculated which will reduce the photons leaving the shield to a predetermined intensity using a minimum quantity of heavy metal. The result for the simple case lends itself to generalization: an inner layer of solid heavy metal may be called for as well as a distribution through the shield. The attenuation of gamma rays in the inner layer should bring the outbound photon current down to the value found by multiplying the source strength (cm^{-3} sec^{-1}) of outbound photons just outside the layer by half the radius. The distribution of heavy metal through the slowing down portion of the shield should be that, which, in the absence of photon sources, would cause a photon current attenuation equal to the attenuation of the actual photon source. It is probable that neutron and photon tolerances

being what they are, it is not necessary to have the heavy metal loading extend through to the outside of the shield. (auth)

9894 NAA-SR-942
North American Aviation, Inc., Downey, Calif.
HEAT GENERATION IN THERMAL SHIELDS. M. Heisler and J. Wetch. Mar. 26, 1954. Decl. Mar. 7, 1957. 45p. Contract AT-11-1-GEN-8. $0.35(OTS).

Heat production resulting from the absorption of γ-ray photons in thermal shields and the leakage of neutrons and photons from ferritic thermal shields are investigated. The γ rays considered arise from three types of reactor radiation—thermal neutrons, fast neutrons, and core and reflector gammas. The energy spectra of the fast neutron leakage and absorption have been investigated in some detail because of the significant contribution of fast neutrons to the heating of the concrete biological shield. As far as the thermal neutron capture gammas and the core and reflector gammas are concerned, the data presented here are general and can be applied to any core and reflector geometry whose diameter is large compared with the thermal shield thickness. This is not true for the fast neutron capture gammas, since their attenuation depends upon the energy spectrum of the incident neutron current. The data given in this report were calculated on the basis of resolving the spectrum emerging from a 61-centimeter graphite reflector into a number of monoenergetic plane sources. In the regions where the reflector is much thicker than this, (i.e., at edges and corners) there is so little fast neutron current emerging, and even this is largely thermalized, that the shield heating calculations can proceed as though all the incident neutrons are thermal. (auth)

Refer also to abstract 9782.

SPECTROSCOPY

9895 CF-56-8-168
Oak Ridge National Lab., Tenn.
INFRARED PROGRAM OF THE ORNL SPECTROSCOPY RESEARCH LABORATORY. H. W. Morgan, P. A. Staats, J. H. Goldstein, and J. R. McNally, Jr. Aug. 24, 1956. 8p. Contract [W-7405-eng-26]. $1.80(ph OTS); $1.80(mf OTS).

A brief survey is made of the research on infrared spectroscopy at ORNL. (B.J.H.)

9896 CF-56-9-126
Oak Ridge National Lab., Tenn.
PRELIMINARY INVESTIGATION OF THE DIFFERENTIAL INFRARED SPECTRA TECHNIQUE FOR CHARACTERIZING THE SURFACE OF THORIUM OXIDE. M. A. Tamers. Sept. 14, 1956. 7p. Contract [W-7405-eng-26]. $1.80 (ph OTS); $1.80(mf OTS).

URANIUM AND URANIUM COMPOUNDS

9897 A-41
Minnesota. Univ., Minneapolis.
THE DETERMINATION OF THE COEFFICIENT OF THERMAL DIFFUSION OF URANIUM HEXAFLUORIDE. Alfred O. Nier. May 28, 1941. Decl. Jan. 30, 1957. 9p. $1.80 (ph OTS); $1.80(mf OTS).

By means of a Clusius type thermal diffusion column, an attempt has been made to evaluate the coefficient of thermal diffusion of UF_6 in the temperature range between 65 and 480°C. As no appreciable separation was observed, it is concluded that this coefficient is so small that the method could not be used in this range for the separation of quantities of U^{235}. (auth)

9898 ANL-5070
Argonne National Lab., Lemont, Ill.
FAST FISSIONS IN NATURAL URANIUM RODS. Samuel Untermyer. June 12, 1953. Decl. Feb. 12, 1957. 12p. Contract W-31-109-Eng-38. $0.25(OTS).

The ratio (δ) of fast fissions to thermal fissions has been measured for rods up to 4 inches in diameter. Measurements were made with a $\frac{1}{4}$-in. diameter fission ion current chamber, placed at various positions within the rods. These values are in good agreement with the measurement of D. L. Hill, as reported in CF-1548. The value of the fast effect (ϵ) is approximately $1 + \delta/2$. (auth)

9899 HW-34868 C 8
Hanford Atomic Products Operation, Richland, Wash.
MECHANICAL AND PHYSICAL PROPERTIES OF URANIUM. Chapter 8 of NUCLEAR METALLURGY LECTURES. R. E. Hueschen. May 10, 1955. Decl. Mar. 13, 1957. 32p. Contract [W-31-109-Eng-52]. $0.35(OTS).

The physical properties such as density, thermal and electrical conductivity, and thermal expansion and the mechanical properties such as tensile and elastic behavior, fatigue, hardness, shear strength, creep, and impact for pre-irradiated U are given. (F.S.)

9900 KAPL-1044
Knolls Atomic Power Lab., Schenectady, N. Y.
THE DIMENSIONAL STABILITY OF URANIUM SINGLE CRYSTALS AND POLYCRYSTALS DURING IRRADIATION AND THERMAL CYCLING. A. M. Turkalo. Feb. 8, 1954. Decl. Feb. 25, 1957. 15p. Contract W-31-109-Eng-52. $0.25(OTS).

The results of a number of irradiation and cycling tests performed with unalloyed uranium during the past year and a half are summarized. Single crystals that show no dimensional change after thermal cycling exhibit considerable change after irradiation. Elongation occurs in the [010] direction and shortening in the [100] direction. Polycrystalline uranium is dimensionally unstable during irradiation in a fashion dependent on the preferred orientation present in the rod and consistent with the single crystal irradiation results. Although thermal cycling of uranium leads to distortion resembling that produced in the pile and depends on some of the same variables, there is not a one to one correlation between thermal cycling and irradiation results. (auth)

9901 ORNL-2011
Oak Ridge National Lab., Tenn.
THE TECHNOLOGY OF URANIUM DIOXIDE—A REACTOR MATERIAL. J. R. Johnson, L. M. Doney, S. D. Fulkerson, A. J. Taylor, J. M. Warde, and G. D. White. Feb. 21, 1956. Decl. Mar. 2, 1957. 34p. $0.40(OTS).

Interest in the use of UO_2 as a reactor fuel resulted in a detailed study of its chemical and physical properties. The crystalline structure, thermal properties, oxidation behavior, chemical activity, methods of preparation and analysis, and susceptibility to radiation have been investigated. Results are reported. (D.E.B.)

9902 NAA-SR-Memo-722
North American Aviation, Inc., Downey, Calif.
LOADING SCHEDULE AND CONTROL REQUIREMENTS OF THE U^{233} PRODUCTION UPR CORE. Ralph Balent. June 22, 1953. Decl. Feb. 27, 1957. 15p. Contract [AT-11-1-GEN-8]. $3.30(ph OTS); $2.40(mf OTS).

9903 UCRL-1620
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